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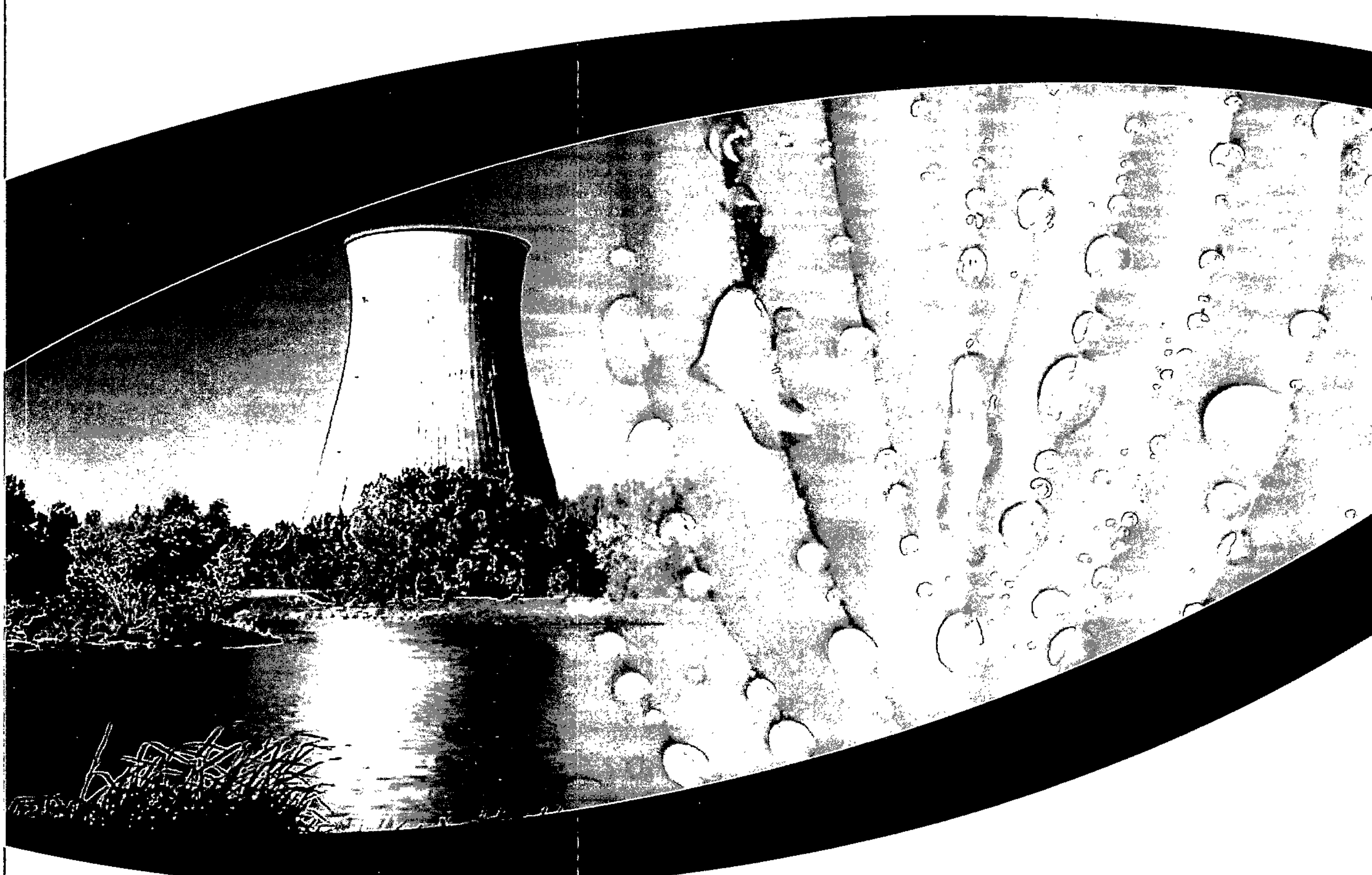
2011 TECHNICAL REPORT

Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines—Revision 4



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Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines—Revision 4

1022832

Final Report, November 2011

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This document does **NOT** meet the requirements of 10CFR50 Appendix B, 10CFR Part 21, ANSI N45.2-1977 and/or the intent of ISO-9001 (1994)

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ACKNOWLEDGMENTS

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This publication is a corporate document that should be cited in the literature in the following manner:

Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines—Revision 4. EPRI, Palo Alto, CA: 2011. 1022832.

REPORT SUMMARY

Primary-to-secondary leakage of steam generator tubes in pressurized water reactors (PWRs) can result from mechanisms that propagate slowly or rapidly. Control room operators rely on online data for a rapid assessment of tube leakage conditions to ensure that the plant is maneuvered safely and to minimize the risk of tube rupture.

Industry experts prepared and reviewed these revised guidelines to incorporate recent industry operating experience and technology improvements and to review the technical bases for action levels. This report represents Revision 4 of the industry-wide guidelines originally issued in 1995 to address the issue of primary-to-secondary tube leakage.

Background

This is the fourth revision to the primary-to-secondary leak rate guidelines. The original recommendations, published in the Electric Power Research Institute (EPRI) report *PWR Primary-to-Secondary Leak Guidelines* (TR-104788) in May 1995, were developed by a committee of industry experts from a variety of disciplines under the auspices of EPRI. The guidelines were updated in 1997 (Revision 1; TR-104788-R1) and again in 2000 (Revision 2; TR-104788-R2) to review and update the guidelines to reflect changes in technology, understanding of tube material performance, and operating experience. Revision 3, published in December 2004 as EPRI report 1008219, focused on updating industry experience and program development consistent with Steam Generator Management Program (SGMP) guidance.

These guidelines are designed to provide utility personnel with a defense-in-depth approach to ensuring that operating with low-level primary-to-secondary leak rate has a low probability of escalating to a tube rupture event.

Objective

To provide operators with guidance related to operational responses, management considerations, monitoring methods and equipment, leak rate calculations, and data evaluation for the following scenarios:

- Low-level or slowly increasing primary-to-secondary leak rate
- Rapidly increasing primary-to-secondary leak rate
- Steam generator tube rupture (no leak before break)

Approach

The guidelines committee—composed of utility personnel, contractors specializing in steam generator tube and material issues, and EPRI personnel—reviewed the technical bases for Section 2, the operational limits and actions described in Section 3, and recent operating experience with *PWR Primary-to-Secondary Leak Guidelines–Revision 3*. The committee divided into multiple subcommittees to evaluate and revise the various sections of the guidelines

to incorporate updates based on human performance standards tools and operating experiences. The recommended changes are intended to clarify implementation of the guidelines and to ensure that the likelihood of propagation of flaws to tube rupture is minimized under both normal and faulted conditions. These improvements are designed to aid utilities in the implementation of the guidelines.

Results

The committee revised the guidelines consistent with the SGMP Administrative Guideline and previous guideline formats. The revision continues the designation of "mandatory" and "shall" requirements, consistent with the implementation philosophy of Nuclear Energy Institute (NEI) 97-06, *Steam Generator Program Guidelines*, and clarified with NEI 03-08, *Guideline for the Management of Materials Issues*.

These guidelines capture two methodologies for detecting and monitoring leak rate—rate of change and fixed leak rate criteria. In both cases, conditions can result in rapid, controlled shutdown if primary-to-secondary leak rate displays evidence of undesirable propagation. Four different operating conditions are defined and better aligned with plant operating conditions related to heat-up, cooldown, normal operation, and transient conditions.

Keywords
Chemistry control
Corrosion control
Plant maintenance assistance
Primary-to-secondary leak rate
Radiation monitoring
Steam Generator Management Program

ABSTRACT

Primary-to-secondary leak rate can result from mechanisms that propagate slowly or rapidly. The willingness of the industry to prescribe and implement operational responses to primary-to-secondary leak rate, as described in the *PWR Primary-to-Secondary Leak Guidelines*, has contributed to the successful minimization of tube rupture events since 1995. The committee most recently reconvened in 2009 and 2010 for Revision 4 to these guidelines. The committee was a consortium of industry experts, utility personnel, and contractors focused on reviewing current operating experience, research and development activities from the Steam Generator Management Program, and human performance tools developed in the management and operation of plants with primary-to-secondary leak rate. As established in earlier versions, these guidelines provide stations with an effective program for monitoring, operating, and responding to primary-to-secondary leak rate under normal and faulted conditions. These guidelines contain "recommended," "shall," and "mandatory" operating requirements of Nuclear Energy Institute (NEI) 97-06, *Steam Generator Program Guidelines*, and NEI 03-08, *Guideline for the Management of Materials Issues*, in response to primary-to-secondary leak rates of varying magnitudes, methodologies for evaluating them, and action for changes in the leak rate. These guidelines continue the emphasis on inline radiation monitors for control room operators to enable a rapid detection and response to increasing leakage due to rapidly propagating tube failure events.

FOREWORD

The EPRI steam generator guidelines, including the *PWR Primary-to-Secondary Leak Guidelines*, focus on maximizing equipment reliability and assessments and providing a defense-in-depth approach to minimizing the potential tube rupture. This also provides utility personnel with the ability to safely operate with and respond to low-level primary-to-secondary leak rate events. Implementation of these guidelines continues to contribute to the successful defense against steam generator tube rupture events.

Since their initial release in 1995, these guidelines have undergone three revisions, with a focus on implementing a sound, technically based monitoring program and continuously updating actions and methodologies based on industry operating experiences. The 1997 and 2000 revisions focused on laboratory data and industry operating experience, whereas the 2004 revision was updated to reflect Steam Generator Management Program (SGMP) expectations. Revision 4 focused on updating the technical bases and clarifying the monitoring and action-level requirements for implementation, based on the lessons learned from Revision 3.

EPRI formed the Revision 4 committee, consisting of industry experts, utility members, contractors, and EPRI personnel. EPRI hosted four meetings from 2009 through 2010 to review and revise the guidelines. Some of the changes incorporated were the following:

- Many of the "mandatory" and "shall" requirements have been reworded for clarity. In general, these changes are more specific, but the original intent has not changed.
- A number of recommendations in the text that were somewhat uncertain with respect to whether they were NEI recommendations were specifically defined as NEI recommendations.
- Section 2 was rewritten for clarity with relatively little re-analysis. Sections specifically addressing other industry guidance on primary-to-secondary leak rate were added.
- Section 3 was reorganized. One major element of this reorganization was to provide an explicit alternative program for monitoring only the leak rate (that is, not monitoring the rate of change in leak rate), which had been defined in Revision 3 only through using a leak rate limit as an alternative indicator of rate of change in leak rate.
- Refinement of SGMP Administrative Procedure actions into these guidelines was continued. Section 6 is new, based on recent SGMP guidance documents identifying the "mandatory," "shall," and "recommended" requirements. Section 3 contains the operating conditions, monitoring, and action level requirements. "**<u>Mandatory" elements are highlighted in bold and underlined and indicated as "mandatory" requirements.</u>** **"Shall" requirements are highlighted in bold and indicated as "shall" requirements.**

- Section 3.3 states that each plant should define the leak rate methodology. Sites are to specify whether the Primary-to-Secondary Leak Rate program will use the rate-of-change or fixed leak rate methodologies. It is expected that sites do not change or alter methodologies during an active event. This is not a "mandatory" or "shall" requirement.
- Mode 3 and 4 monitoring requirements were revised. In Revision 3, the Mode 3 and 4 requirements were "mandatory," and the Revision 4 committee changed this to a "shall" requirement to meet Technical Specifications.
- Table 3-1 was revised and requires a faster shutdown in Revision 4 if leakage is >75 gpd and there are no continuous radiation monitors.
- Action Level 1 entry requirements are defined as "shall" requirements compared to the Revision 3 "mandatory" requirements.
- Revision 4 clarified the Action Level 3, constant leak rate methodology requirement to commence shutdown and be at < 50% power in one hour and at mode 3 in three hours.
- Revision 4 clarified that the constant leak rate methodology applies to the confirmed leak rate spikes (not electronic signals).
- Section 4 was reviewed, with minor editorial revisions.
- Section 5 received minor edits, primarily focusing on adding clarity to the cautions associated with each of the equations for calculation of leak rate.
- Section 6 was added to specifically delineate NEI 03-08 guidance elements, consistent with other SGMP documents.
- Appendix A was added to document recent observations on industry experience with the effect of primary-side pH on primary-to-secondary leak rate.
- Appendices B, C, and D were edited to provide additional cautions regarding the equations presented, correct minor typographical errors, and provide consistent formatting of equations.
- Appendix E was added to provide a description of industry experience with primary-side argon injection.

Station personnel implementing the program provide an integral component to the defense-in-depth approach. A comprehensive program consisting of training, qualification, administration, and procedures provides utilities with a sound technical program designed to minimize the likelihood of a tube rupture event. These guidelines do not ensure that a tube rupture will not occur, but they provide utilities with the ability to manage low-level primary-to-secondary leak rate and continue the successful implementation that has been taking place since the inception of the guidelines.

ACRONYMS, ABBREVIATIONS, AND TERMINOLOGY

This section defines acronyms, abbreviations, and special terms used in this report.

ABB/CE	ABB Combustion Engineering
activity	concentration of radioactivity in liquid (μCi/g or μCi/ml, as appropriate) or gas (μCi/cc)
AE	air ejector
ALARA	as low as reasonably achievable
ANO	Arkansas Nuclear One
ARC	alternative repair criteria
ASME	American Society of Mechanical Engineers
BAT	boric acid treatment
BOP	balance-of-plant
B&W	Babcock & Wilcox
CE	Combustion Engineering
CFR	Code of Federal Regulations
COG	condenser off-gas
CPM	counts per minute
faulted condition	design basis accident condition such as a steam line break
GM	Geiger-Muller
GRW	gaseous radioactive waste system
HEPA	high-efficiency particulate air
HOH	hydrogen ion; hydroxyl ion form resin
ID	inner diameter
IGA	intergranular attack
LLD	lower limit of detection
MCA	multiple channel analyzer
MSLB	main steam line beak

NDE	nondestructive examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
ODSCC	outer diameter stress corrosion cracking
OTSG	once-through steam generators
P_b	burst differential pressure
P_{MSLB}	differential pressure for main steam line break
PSL	primary-to-secondary leak
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
RCS	reactor coolant system
RMS	radiation monitoring system
RPC	rotating pancake coil
RSG	recirculating steam generator
SCC	stress corrosion cracking
SG or S/G	steam generator
SGMP	Steam Generator Management Program
SGTR	steam generator tube rupture
SOER	significant operating experience report
STP	South Texas Project
TBAOH	tetrabutylammonium hydroxide
TSTF	Technical Specification Task Force
UFSAR	updated final safety analysis report
USNRC	United States Nuclear Regulatory Commission
VCT	volume control tank
WE	Westinghouse

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1 INTRODUCTION AND MANAGEMENT RESPONSIBILITIES

1.1 Background

Historically, steam generator tubes in pressurized water reactors (PWRs) have experienced various types of degradation from both the primary and secondary sides of the tubes. Corrosion mechanisms of concern include wastage, pitting, outer diameter stress corrosion cracking (ODSCC), intergranular attack (IGA), and primary water stress corrosion cracking (PWSCC). Mechanical damage from fretting, fatigue and foreign objects has also resulted in tube degradation.

Utility inspection and diagnostic programs are designed to detect incipient conditions before steam generator tube corrosion or mechanical damage lead to through-wall failure. In most cases, tube degradation mechanisms that result in primary-to-secondary leakage propagate slowly and can lead to operational difficulties, but do not diminish safety margins. However, some damage mechanisms can progress rapidly and can result in a tube rupture, resulting in significant secondary system contamination and potential actuation of reactor safety systems.

The EPRI *Pressurized Water Reactor Steam Generator Examination Guidelines* establishes a recommended practice for NDE that reflects utility operating experience and research and development efforts [1]. Tubes are repaired or removed from service when inspection data indicate that their integrity is no longer assured [2].

The purpose of the *Pressurized Water Reactor Primary-to-Secondary Leak Guidelines* is to reduce the probability of a steam generator tube rupture. These *Guidelines* do not directly address the issue of steam generator structural or accident leakage criteria (e.g., structural integrity at three times the nominal operating pressure differential and the ability to limit accident leakage within specifications, typically 1 gpm). Operability and Technical Specification compliance are factors in the administrative leakage limits and action statements of these *Guidelines*. Tube integrity is addressed in other EPRI guidance [1, 2].

The shutdown leakage limits and responses to leakage presented in Section 3 have been developed within the context of defining responses to primary-to-secondary leaks as a *defense-in-depth* measure. Specifically, the bases for the operating limits are such that adherence to these limits will significantly reduce the already low likelihood of a steam generator tube rupture event.

This revision to these *Guidelines* incorporates the experience gained since Revision 3 was issued in December 2004.

1.2 Guidelines Objectives

The objective of these *Guidelines* is to provide operators with guidance related to operational responses, management considerations, monitoring methods and equipment, leak rate calculations, and data evaluation for the following scenarios:

- Low-level and/or slowly increasing primary-to-secondary leakage
- Rapidly increasing primary-to-secondary leakage
- Steam generator tube rupture (no leak before break)

1.2.1 Guidelines Framework

In 1997, the U.S. nuclear power industry established a framework for increasing the reliability of steam generators by adopting NEI 97-06, *Steam Generator Program Guidelines* [3–6]. This initiative references EPRI's *Pressurized Water Reactor Primary-to-Secondary Leak Guidelines* as the basis for an industry consensus approach to primary-to-secondary leak programs. Specifically, NEI 97-06 requires that U.S. utilities meet the intent of the EPRI *Pressurized Water Reactor Primary-to-Secondary Leak Guidelines*.

The U.S. nuclear power industry has more recently (May 2003) produced a policy that commits each nuclear utility to adopt the responsibilities and processes on the management of material aging issues described in NEI 03-08, *Guideline for the Management of Materials Issues* [7–9]. The scope of NEI 03-08 extends to:

- "PWR and BWR reactor vessel, reactor internals, and primary pressure boundary components"
- "PWR steam generators (SG)"
- "Non Destructive Examination (NDE) and chemistry/corrosion control programs that provide support to the focused programs above"

In addition, NEI 03-08 states, "as deliverables or guidelines are developed, expected actions should be classified as to relative level of importance:"

- "Mandatory—to be implemented at all plants where applicable"
- "Needed—to be implemented whenever possible but alternative approaches are acceptable"
- "Good practice—implementation is expected to provide significant operational and reliability benefits, but the extent of use is at the discretion of the individual plant/utility"

These *Guidelines* provide utility personnel with adequate detail for the development of procedures for operators to respond to changes in steam generator tube leakage in a safe, reliable and cost-effective manner. Implementation of these *Guidelines* will improve the monitoring of small leaks and reduce the potential for steam generator tube ruptures.

These *Guidelines* present an industry-developed approach for calculating and monitoring primary-to-secondary leakage. These *Guidelines* do not address accident and operational leakage limits which are addressed in the plant Technical Specifications as amended by TSTF 449 Revision 4 [10].

1.2.2 Steam Generator Management Program Categories

The *Steam Generator Management Program (SGMP) Administrative Procedures, Revision 3*, issued in December 2010, gives guidance that aligns the SGMP with NEI 03-08, but with slightly different terminology for the categories of guideline elements [11]. These categories are described as follows:

Mandatory: Guideline elements designated as "Mandatory" are to be implemented at all plants where applicable. Each utility is ultimately responsible for the operation of their plant(s) and actions taken at those plants, but should realize that it is highly unlikely that any deviations from "Mandatory" elements would be supported by the industry. If a situation develops that was not considered during development of the industry guideline and a utility decides to deviate from a "Mandatory" requirement, the NEI-03-08 deviation process shall be used (Section 2.12 of [11]).

Per NEI 03-08, criteria that qualify an element of a guideline as "Mandatory" include:

1. Element substantively affects the ability of structures, systems and components to perform their intended safety function.
2. Element would be highly risk significant as determined by the applicable TAC if not implemented.
3. Element poses a significant threat to continued operation of the affected plants, including economic threats that could reasonably lead to protracted plant shutdown or retirement.
4. A consensus of the IC believes the element should be designated as "Mandatory".

<u>**Mandatory requirements are in bold and underlined in this document.**</u>

Shall (equivalent to "Needed" elements in NEI 03-08): Guideline elements designated as "Shall" are to be implemented wherever possible, but alternative approaches are acceptable. If a utility decides to deviate from a "Shall" requirement NEI-03-08 deviation process shall be used (Section 2.12 of [11]).

Per NEI 03-08, criteria that qualify an element of a guideline as "Needed" or "Shall" include:

- Element substantively affects the ability of structures, systems or components to reliably perform their economic function.
- Element would be moderately risk significant as determined by the applicable TAC if not implemented.
- Element addresses a material degradation mechanism that has significant financial impact on the entire industry, especially where failure at one plant could affect many other plants.
- A consensus of the IC believes the element should be designated as "Shall".

Shall requirements are in bold in this document.

Recommendations (equivalent to "Good Practice" elements in NEI 03-08): Guideline elements, designated as "Recommendations," are expected to provide significant operational and reliability benefits, but the extent of use is at the discretion of the individual utility. Written documentation is not required when a "Recommendation" is not implemented.

Per NEI 03-08, criteria that qualify an element of a guideline as "Recommended" include:

- Element reflects an industry standard of performance or represents a consensus opinion of the applicable TAC.
- A consensus of the IC believes the element should be designated as a "Recommendation".

These *Guidelines* contains Mandatory Requirements and are identified as **Mandatory Entry Requirements** and **Mandatory Requirements**. These *Guidelines* contain Shall Requirements and are identified as **Shall Entry Requirements** and **Shall Requirements**.

Entry Requirements define the plant conditions that result in entry into a specific monitoring requirement(s) or Action Level. Requirements define the specific monitoring requirement or actions in response to Entry Requirements.

This document contains Mandatory, Shall, and Recommended elements. These requirements are presented in Section 3 and summarized in Section 6. All other parts of these *Guidelines* are to be considered informational only.

1.3 Primary-to-Secondary Leak Program Considerations

This section discusses the considerations common to most utilities, including the elements of organizations that are needed to effectively carry out the primary-to-secondary leak monitoring program. Utility-specific implementation policies and procedures typically assign the responsibilities to specific positions within the organization. It is imperative that all levels of utility management understand the importance of the actions presented in Section 3 and the potential impact on, and benefits to, the utility company.

Control Room operator's actions are driven based on radiation monitors readings. Accordingly, increased emphasis is to be placed on the reliability of these monitors. Leakage, as determined from radiation monitor data, is based on a correlation between activity and leakage. Since the activity of the primary side can change over time (for example, a fuel leak), periodic grab sample results are generally used to update the correlations between monitor response and leakage that are used to determine the level of leakage. This concept is important in implementing an integrated leakage program since grab sample results are time-consuming and cannot provide timely data if degradation is rapidly progressing. These *Guidelines* provide the recommended methodology for calculating leakage from grab samples but emphasize the importance of relying on continuous readings from system radiation monitors.

It is important that plant operators recognize that radiation monitors for different systems, or different parts of the same system, will not have the same response time from the onset of a leakage event. Also, leakage in different locations will lead to different response times from various monitors. For example, during power operation leakage would be detected by an N-16 monitor within a few seconds of an event whereas the condenser offgas radiation monitor may have a time delay of several minutes, and the blowdown sample may have indications delayed as much as 1 hour. The time delays for responses from these monitors will typically be evaluated by plant personnel and reflected in plant procedures and training programs. This will ensure that responses occur within appropriate time frames and will enable operators to assess complementary radiation monitor responses in the proper time frames. This also applies to changing leakage and correlating grab sample results for radiation monitor readings. In many cases the grab sample location and the radiation monitor analysis point are not the same. Any time delay differences need to be factored into this correlation when assessing radiation monitor response factors.

1.3.1 Program

One aspect of a successful primary-to-secondary leak monitoring program is a well developed set of procedures that implement these *Guidelines*. Program documents may:

1. State the need for the program
2. Assign responsibility for:
   - Preparing and approving procedures to implement the program
   - Monitoring, analyzing, and evaluating data for the monitoring program
   - Establishing priorities for equipment/instrument maintenance
   - Reviewing program functions
   - Providing corrective actions prescribed by the program
   - Ensuring continued operation with low-level leakage is consistent with the recommendations contained in these Guidelines and with plant Technical Specifications
   - Conducting walk-through scenarios or drills to test the effectiveness of program response and coordination
3. Establish the authority to:
   - Carry out procedures
   - Implement corrective actions
   - Initiate requests for modifications to plant systems as required to meet the program needs
   - Complete economic analyses
   - Resolve disagreements
   - Initiate plant shutdown if limits are exceeded

When taken together, procedures and other documents implementing this program generally contain the level of detail necessary for personnel at all levels of the utility organization to understand and carry out their responsibilities.

1.3.2 Other Primary-to-Secondary Programmatic Components

1.3.2.1 Contingency Plan for Control and Processing of Large Volumes of Contaminated Water

Plants have prescribed plans for the control and processing of large volumes of contaminated water in the secondary side of the plant. Provisions are established to confine the contaminated water in the condenser hotwell. Capabilities to transfer excess water to another storage area prior to exceeding hotwell capacity are typically addressed. Necessary contingencies for storage areas where contaminated water can be drained are typically considered in a prescribed sequence. Contingency plans for processing large volumes of contaminated water are typically established.

1.3.2.2 Other Recovery Activities

There are many station activities that result from primary-to-secondary leakage. These include addressing radiological consequences resulting from balance of plant (BOP) contamination, rapid outage scheduling resulting from unplanned shutdown, radioactive waste processing of large water volumes, etc. Recommendations concerning these activities are out of the scope of these *Guidelines* and are not addressed. However, the importance of contingency plans for these activities is not to be overlooked by station management.

1.3.2.3 Response to Confirmed Primary-to-Secondary Leakage

When primary-to-secondary leakage is first confirmed at a plant, management will typically review the overall program for dealing with primary-to-secondary leaks, including the elements covered above. Additionally:

1. Systems in the secondary plant are expected to be placed in conditions to minimize the spread of contamination by pathways such as:
   - Turbine building sump effluent paths
   - Hotwell return to condensate storage tanks
   - Condenser off gas system (e.g., realign through high efficiency particulate air (HEPA)/charcoal filter system if available)
   - Use of condensate polishing resin beds, which will remove radionuclides from the secondary water
2. Plant resources should be reviewed and additional resources provided, if needed, such as:
   - Operations staffing
   - Chemistry staffing
   - Radiological Control staffing
   - Water processing capability
   - Makeup water capability
   - Secondary contamination and containment capabilities

1.4 Training

1.4.1 General

Utilities typically provide for periodic (continuing) training of personnel implementing the program commitments. Indoctrination in the basics of the program is expected to be considered for all employees who, by virtue of their job responsibilities, may identify and respond to symptoms of primary-to-secondary leakage.

Training programs are generally designed for the level and qualifications of personnel being trained. The following elements will typically be included:

- A clear statement of the primary-to-secondary leak policy, including clarification of the impact of this policy on the various areas of responsibility
- Identification of the relationship between primary-to-secondary leakage monitoring and commitments to primary system integrity and off-site dose calculations
- Techniques/methodology used for monitoring leakage and leakage rate of change and
- The interaction/communication required between station personnel to ensure the commitments of the program are satisfied

1.4.2 Plant Simulator Training

Operators will typically be trained to monitor for primary-to-secondary leakage and to take actions in response to an increasing tube leakage trend.

Simulator training scenarios are generally meant to be representative of industry experience and to include one model wherein all decisions are based on radiation monitoring responses to demonstrate usage and compliance with these *Guidelines*. The usefulness of simulator training is increased if scenarios are varied to include different radiation monitor responses (from different radiation monitors), which would initiate the operational actions contained in these *Guidelines*.

1.5 General Guidelines for Development of a Primary-to-Secondary Leak Administrative Program

The following general guidelines are offered to show an acceptable approach for development of an administrative program for management of primary-to-secondary leakage.

1. Evaluate the plant Technical Specification requirements for instantaneous concentrations (both liquid and gaseous effluents) and the off-site accumulative doses for all site-specific pathways. This evaluation is expected to incorporate station ALARA goals to maintain off-site doses well below those required by 10CFR20 and 10CFR50 Appendix I.
2. Evaluate the present plant storage and processing capabilities (conventional waste systems) for contaminated secondary system water and demineralizer resins, if applicable, for various primary-to-secondary leakage scenarios to assess the plant's ability to meet the regulatory effluent requirements. These scenarios include various leak sizes (small, slowly propagating/steady leaks, up to a tube rupture) for various RCS activity source terms. If the waste system storage and/or treatment capabilities do not meet the plant effluent requirements and management goals, then develop an action plan to upgrade the systems.

3. Define the bases and limitations of the primary-to-secondary leak program and the bases related to the use of the rate of change methodology or constant leakage methodology. These *Guidelines* are designed to allow utilities the option of either methodology but the bases and decision process for operators are expected to be documented.
4. Consider administrative leakage limits to minimize the likelihood of exceeding effluent radioactivity release rates and generating widespread plant contamination. These limits can be used to initiate prompt leakage detection and early corrective actions.

1.6 Summary

These *Guidelines* present a generic program for managing primary-to-secondary leakage. Plant-specific programs typically consider plant design, materials, steam generator corrosion experience, foreign object history, management structure, and operating philosophy. However, all plant-specific requirements are expected to be defined in accordance with the intent of these *Guidelines*. This program is expected to assist plant personnel to manage low-level leakage and will reduce the likelihood of tube ruptures. To meet this goal, an effective corporate policy and monitoring program are essential and are generally based on the following:

- Clear management support for operating procedures designed to minimize the likelihood of primary-to-secondary leakage progressing to a tube rupture (Section 3),
- Adequate staff, equipment, and organizational resources to implement an effective leakage monitoring program, using a combination of radiation monitors and laboratory radiochemical analyses,
- A sound leakage monitoring program that incorporates existing plant equipment/radiation monitors and proceduralized actions,
- Management agreement at all levels, prior to implementing the program, on the actions to be taken in response to primary-to-secondary leakage and the methods for resolution of conditions not covered in these *Guidelines*, and
- Continuing review of plant and industry experience and research results to revise the program, as warranted.

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2 TECHNICAL BASES FOR PRIMARY-TO-SECONDARY LEAKAGE LIMITS

2.1 Purpose

The purpose of this section is to provide the technical bases for the primary-to-secondary leakage operating limits and responses defined in Section 3.

The purpose of these *Guidelines* is to reduce the probability of a steam generator tube rupture. A rupture (also referred to as *burst*) is defined in these *Guidelines* as a sudden increase in leakage from a small, manageable level to a large level with significant operational implications. Such an increase in leakage is generally associated with a loss of tube integrity.[1] These *Guidelines* do not directly address the issue of steam generator structural or accident leakage criterion (e.g., structural integrity at three times the nominal operating pressure differential and the ability to limit accident leakage within specifications, typically 1 gpm). Operability and Technical Specification compliance is a factor in the leakage limits and action statements of these guidelines. Tube integrity is addressed in other EPRI guidance [1].

Implementation of these *Guidelines* will not guarantee that tube rupture will be avoided. Mechanisms that might lead to tube ruptures without advanced warning from primary-to-secondary leaks include fatigue cracking at very high stress amplitudes and the breaking of ligaments formed by clusters of stress corrosion cracks. For example, adherence to these *Guidelines* would likely not have prevented the following steam generator tube rupture events:

- Content deleted - EPRI Proprietary
-

The shutdown leakage limits and responses to leakage presented in Section 3 have been developed within the context of defining responses to primary-to-secondary leaks as a *defense-in-depth* measure. Specifically, the bases for the operating limits are such that adherence to these limits will significantly reduce the already low likelihood of a steam generator tube rupture event.

[1] Note that this definition differs slightly from that given in the EPRI *Integrity Assessment Guidelines*, which explicitly define a burst to include "gross structural failure" of the tube wall. The difference in definition will generally only be relevant for leaks from flaws that are not cracks. This is discussed further in Section 2.2.3.1.

The remainder of this section is organized into the following subsections:

- Section 2.2 discusses the technical bases relating tube rupture probabilities to leakage. Two different flaw types are addressed. Section 2.2.1 discusses stress corrosion cracks. Section 2.2.2 discusses fatigue cracks. Other flaw types are discussed in Section 2.2.3, including wear due to foreign objects and support structures (Section 2.2.3.1).
- Section 2.3 discusses issues which are not considered in the models developed in Section 2.2, but may affect the relationship between measured leakage and the probability of tube rupture.
- Section 2.4 reviews industry guidance documents which have been considered in developing the technical bases present in these *Guidelines*.
- Section 2.5 lists the references used in developing these technical bases.

This section discusses the relationships between tube flaws and primary-to-secondary leakage and rate of increase in leakage. Actions taken by utilities using these relationships require determination of leakage with reasonable accuracy. Methodologies for determining leakage (and through multiple determinations, the rate of increase in leakage) are discussed in depth in Section 5. This section assumes that a leak rate has been appropriately calculated. The technical bases developed in this section have generally been based on probabilistic assessments. Typical uncertainties in the measurement of primary-to-secondary leaks are thought to be much less than those already incorporated into the analyses given in this section. Therefore, when implementing shutdown criteria developed from the technical bases presented in this section, it is not necessary to adjust for typical uncertainties in the measured leakage value; a best estimate is sufficient. Additional detail regarding evaluation of uncertainties can be found in References [4] and [5].

2.2 Technical Bases for Leakage Limits

2.2.1 Technical Bases for Limits on Leakage from Stress Corrosion Cracks

This section discusses the technical bases for limits on the instantaneous leakage from a stress corrosion crack, i.e., the rate at which primary coolant flows through a crack into the secondary side at any given time. Section 2.2.1.1 describes the methodology for determining leak rate limits. Sections 2.2.1.2, 2.2.1.3, and 2.2.1.4 give the detailed analyses of that methodology. In Section 2.2.1.5 selected operating experience is compared to the results of the analyses discussed in Sections 2.2.1.2, 2.2.1.3, and 2.2.1.4 to demonstrate its appropriateness.

2.2.1.1 Assessment Methodology

The assessment methodology for establishing a leakage limit for stress corrosion cracks consists of the following steps:

- Determining the structurally limiting flaw size (Section 2.2.1.2)
- Determining leakage as a function of flaw size (Section 2.2.1.3)
- Combining the results of the first two steps to determine the structurally limiting leakage (Section 2.2.1.4)

2.2.1.2 Structurally Limiting Flaw Size

To determine the size of a structurally limiting flaw in a steam generator tube, it is necessary to know the required minimum burst pressure. The minimum burst pressure is the minimum differential pressure across the tube wall which will lead to gross structural failure of the tube wall and is typically understood as the pressure differential required to make a given flaw opening unstable (i.e., the flaw size will increase under a constant pressure differential). As Content deleted - EPRI Proprietary

In general, the minimum acceptable burst pressure for steam generator tubes for condition monitoring or operational assessment [1] is taken as the greater of:
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As discussed in Section 2.1, these *Guidelines* are a defense-in-depth measure and are meant only to reduce rather than eliminate the likelihood of a tube rupture. The minimum acceptable burst Content deleted - EPRI Proprietary

Once a minimum burst pressure has been determined, the structurally limiting flaw size may be determined. Westinghouse performed a series of calculations for EPRI during the Revision 2 process relating burst pressure to flaw size using the EPRI through-wall crack burst correlation ([6]). The input parameters to this calculation were geometry factors (tube diameter and wall thickness) and material properties (yield stress, S_y, the sum of the yield and ultimate stress, S_y+S_u, and the standard deviation in the sum of the yield and ultimate stress, σ_{yu}) as shown in Table 2-1. The materials properties used were for an elevated temperature (e.g. 620-650°F).

The results of the Westinghouse calculations are given in Figures 2-1 and 2-2, which give the nominal results and the 95th percentile results of a Monte Carlo analysis, respectively. The nominal and 95th percentile results are shown only to demonstrate the range of results. Note that the correlation between leakage and burst pressure differential developed in Section 2.2.1.4 uses the full distribution of the results.

During the Revision 4 process, an independent check [8] of the nominal burst pressure calculations was performed using the Zahoor correlation (a standard analysis for tube rupture, see Reference [9], selected on the basis of convenience, since this check was only for reasonableness). This check indicated that the results presented in Figure 2-1 are reasonable.

Table 2-1
Geometry, Material Properties(1) and Normal Operation Parameters for the SG Tubes Analyzed

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Figure 2-1
Nominal Throughwall Crack Burst Pressure vs. Crack Length: Material Properties Alloy 600 SG Tubes at 650°F

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Figure 2-2
Lower 95th Percentile Throughwall Crack Burst Pressure vs. Crack Length: Material Properties Alloy 600 SG Tubes at 650°F

2.2.1.3 Leakage as a Function of Flaw Size for Stress Corrosion Cracks

This section presents the methodology used to determine the relationship between the flaw (stress corrosion crack) size and the leakage. As a bounding crack type, Content deleted - EPRI Proprietary

Numerical codes such as CRACKFLO (Westinghouse), PICEP (EPRI) [11], and SQUIRT (NRC) can be used to establish a relationship between flaw size and leakage. These codes each use a two-phase flow algorithm to predict the mass flow rate from a crack of a given length based on the pressure differential across the crack, the temperature, and the mechanical properties of the tube. During leakage, the pressure drop across the leak may lead to flashing of the primary coolant. These numerical codes can assess such complex flows.

Westinghouse performed a series of calculations for EPRI during the Revision 2 process relating leakage to flaw size for a number of different design conditions using the CRACKFLO code. The input parameters used in this calculation are shown in Table 2-1. The materials properties used were for an elevated temperature (e.g. 620-650°F). As indicated in the table, a number of different tube geometries were analyzed, including tubing sizes used in Westinghouse (WE), Combustion Engineering (CE), and Babcock and Wilcox (B&W) steam generator designs. Different values for the yield stress (S_y), ultimate stress (S_u), and the scatter in the sum of these stresses (σ_{yu}) were used to represent the material differences between designs. As indicated in Table 2-1, different operating pressure differentials were also used for the different designs.

Leakage as a function of crack length for Content deleted - EPRI Proprietary the nominal values given in Table 2-1. The results are given in Figures 2-3, 2-4, and 2-5 for Westinghouse tubes, CE tubes, and B&W tubes, respectively. The 95th percentile results (i.e., the leakage value that is lower than 95% of the calculated results) are also shown in Figures 2-3, 2-4, and 2-5. Note that, as discussed in Section 2.2.1.4, the entire distribution of results was used in a Monte Carlo analysis to develop a relationship between leakage and burst pressure. The nominal and 95th percentile results are included to demonstrate that the differences in leakage arising from design differences is small relative to the possible differences due to uncertainties in the inputs (e.g., material properties, geometric details of the crack, etc.).

During the Revision 4 process, the relationships between flaw size and leakage were evaluated using PICEP [11] as an independent check of the CRACKFLO results. The PICEP calculations were performed assuming Content deleted - EPRI Proprietary and the operating pressure differentials indicated in Table 2-1. Leakage was calculated by PICEP in gallons per minute at 200°F and converted to gpm at STP (standard temperature and pressure). The results for nominal values of the input parameters were within the 95% bounding curve, indicating reasonable agreement. An additional model for leakage as a function of crack length is presented in Figure 9-2 of EPRI's *Integrity Assessment Guidelines* [1]. Laboratory data from SCC specimens are also included in Figure 9-2 of EPRI's *Integrity Assessment Guidelines*. The leakage data from SCC laboratory specimens presented in the *Integrity Assessment Guidelines* (see Figure 9-2 in Reference [1]) are in rough agreement with the results from the CRACKFLO model results given in Figure 2-3 of the present *Guidelines*. These comparisons, along with the PICEP calculation, provide a high degree of confidence in the applicability of the CRACKFLO calculations to the development of the leakage limits.

The discussion in the EPRI's *Integrity Assessment Guidelines* indicates that the model is an upper bound of the laboratory data included in Figure 9-2 of Reference [1]. A comparison between the results from the model in the *Integrity Assessment Guidelines* and the present *Guidelines* shows that the results presented in Figure 2-3 (for example) of the present *Guidelines* are about a factor of five below the leak rate predicted by the model in the *Integrity Assessment Guidelines*. Therefore, using the CRACKFLO model results from Figures 2-3 through 2-5 of the present *Guidelines* to develop the primary-to-secondary leakage limits results in conservative leakage limits.

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Figure 2-3
Normal Operating ODSCC Leak Rates for WE Tubing: Nominal & Lower 95th Percentile

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Figure 2-4
Normal Operating ODSCC Leak Rates for CE SG Tubing: Nominal & Lower 95th Percentile

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Figure 2-5
Normal Operating ODSCC Leak Rates for B&W Tubing: Nominal and Lower 95th Percentile at 50% Confidence

2.2.1.4 Structurally Limiting Leak Rate

The full distribution of correlations between burst pressure and crack length (Section 2.2.1.2) and the full distribution of correlations between leakage and crack length (Section 2.2.1.3) can be statistically combined to predict the normal operating leak rate as a function of the expected burst pressure. The full distribution of correlations includes not just the nominal (that is, the median or 50th percentile) and 95th percentile correlations shown earlier for reference, but all of the correlations of various degrees of confidence.

The two distributions were combined by Westinghouse during Revision 2 using Monte Carlo techniques to obtain the full distribution of correlations between leakage and burst pressure. The resulting nominal and lower 95th percentile leakage/burst curves are presented in Figures 2-6, 2-7, and 2-8. For the correlations between leakage and burst pressure, the nominal curve is the set of pressures for which half the distribution at any given leakage lies above (or below) the burst pressure. This curve is referred to here as a *nominal curve*, but it does not necessarily represent the result of a calculation performed with all of the input parameters at their nominal (or best estimate) values. The resulting trend curves are shown in Figures 2-6, 2-7, and 2-8 for Westinghouse, CE, and B&W tubing, respectively. For a defined burst pressure margin, these curves can be used to obtain the nominal and lower 95th percentile leakage for each tube size. For a given leakage, it is seen that the nominal burst pressures for larger diameter tubing are slightly greater than for smaller diameter tubing. As shown in Figures 2-6, 2-7, and 2-8, the burst pressure at Content deleted - EPRI Proprietary
differential even for the lower 95th percentile values.

Table 2-2 provides the normal operating leakage for each tube size for a crack size predicted to result in tube burst atContent deleted - EPRI Proprietary

As noted above, the principal objective for the leakage limits is to reduce the likelihood for tube ruptures under normal and accident conditions. The goal for the analytical correlations of leakage versus burst pressure given in Figures 2-6, 2-7, and 2-8 is to define the probabilities that shutdown at the recommended leakage would result in burst pressure differentials ΔP_{burst} expected to exceed ΔP_{MSLB} = 2560 psi. These results are given in Table 2-3. For plant shutdown at 75 gpd at STP, the probabilities that the resulting burst pressure differential is greater than ΔP_{MSLB} range from about 87% to 96% for the various tube sizes evaluated. For plant shutdown at 150 gpd at STP, the probabilities that the resulting burst pressure is greater than ΔP_{MSLB} range
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Table 2-2
Normal Operating Leak Rate (gpd) for Flaws Leading to Burst at ΔP_{MSLB} = 2560 psi

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Table 2-3
Probability of $\Delta P_{burst} \geq \Delta P_{MSLB}$ = 2560 psi for Leak Rate (at ΔP_{NO}) of 75 and 150 gpd

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Figure 2-6
Burst Pressure vs. Leak Rate at ΔP_{NO} from an ODSCC Crack: Westinghouse SG Tubing

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Figure 2-7
Burst Pressure vs. Leak Rate at ΔP_{NO} from an ODSCC Crack: CE SG Tubing

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Figure 2-8
Burst Pressure vs. Leak Rate at ΔP_{NO} from an ODSCC Crack: B&W SG Tubing

2.2.1.5 Operating Experience

It is useful to review industry experience with leakage from SCC cracks in order to support the conclusions drawn from the modeling discussed in the previous sections. The measured (or calculated) burst pressure and measured leakage can be compared to one of the correlations given in Figures 2-6, 2-7, and 2-8, providing (with a single data point for each event) some support for the predicted correlation. For this purpose, four primary-to-secondary leak events associated with stress corrosion cracks are reviewed here. These events are as follows:

- ANO Unit 2 (1996)
- McGuire Unit 2 (1997)
- Farley Unit 1 (1998)
- Comanche Peak Unit 1 (2002)

A summary of the data available regarding the leakage events is given in Table 2-4. A summary of the analysis results regarding the crack (leak testing, etc.) is given in Table 2-5. Details of these events are discussed in the sections below. Overall conclusions drawn from the industry experience are given in Section 2.2.1.5.5.

Table 2-4
Stress Corrosion Cracking Events Summary

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Table 2-5
Analysis Summary for Stress Corrosion Crack Leak Rate Events Reviewed

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2.2.1.5.1 ANO Unit 2 (1996)

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Figure 2-9
ANO Unit 2, 1996 Leakage Data (N-16 Monitor)

2.2.1.5.2 McGuire Unit 2 (1997)

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2.2.1.5.3 Farley Unit 1 (1998)

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Figure 2-11
Farley Unit 1, 1998 Leak Rate Data (N-16 Monitor)

2.2.1.5.4 Comanche Peak Unit 1 (2002)

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Figure 2-12
Comanche Peak Unit-1 2002 Leak Rate Experience

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2.2.1.5.5 *Conclusions Regarding Experience with Stress Corrosion Crack Leakage*

The leakage operating experience reviewed in the previous sections indicates that the models used to relate leakage with burst pressure are generally robust, providing reasonable results for different tube geometries, crack locations, and leakage trends (e.g., steady increases versus spikes).

2.2.2 *Technical Bases for Limits on the Rate of Increase in Leakage from Fatigue Cracks*

2.2.2.1 Assessment Methodology

Unlike stress corrosion cracks which tend to grow relatively slowly at a mostly constant rate, fatigue cracks can grow rapidly (relative to plant response times) with accelerating rates. Therefore, in setting limits on leakage, consideration is given here to a limit on the rate of increase in leakage derived from consideration of a fatigue crack. The technical basis for setting leakage increase rate limits is developed using the following steps:

- Stress amplitudes at various locations in different SG designs were determined. (Section 2.2.2.2)
- For a given range of stress amplitudes, crack sizes were predicted as a function of time. (Section 2.2.2.3)
- A fatigue-crack specific correlation between crack size and leakage was developed. (Section 2.2.2.4)
- From the correlation between crack size and leakage, the limiting crack growth rate was converted to a limiting leakage increase rate. (Section 2.2.2.5)

Section 2.2.2.6 discusses operating experience as it relates to these calculated values.

2.2.2.2 Stress Amplitudes

The growth rate of fatigue cracks is dependent upon the stress amplitude experienced at the crack location. Therefore, in order to predict steam generator fatigue crack growth rates it is necessary to have information regarding the stress amplitudes in various locations in various designs.

Table 2-6 provides examples of locations in various steam generators where the stress amplitudes associated with flow induced vibration have been assessed.

Table 2-6
Stress Amplitudes for Steam Generator Tubes

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2.2.2.3 Crack Growth Rates

Circumferential cracks may be initiated by corrosion or fatigue. Cracks from either initiation source may be propagated by fatigue. Fatigue propagation is developed as a result of flow-induced vibration which may be either turbulence or fluidelastic vibration. For circumferential cracks, leakage from vibrating tubes can be expected to be significantly larger than that from static (non-vibrating) tubes, since vibration introduces bending stresses that increase the crack opening for a given flaw size. As the crack angle of a vibrating tube increases, the instability of the tube increases with resulting increases in the vibration amplitude and an associated increase in the rate of crack growth. The crack growth rate can be very high in the approach to severing of the tube (the crack growth rate takeoff).

In the 1980s (for example, Reference [15]) Westinghouse performed a series of calculations to predict the crack growth rates in steam generator tubes associated with fatigue cycling. These calculations were performed for several different stress amplitudes and plant-specific temperatures. The results are shown in Figure 2-13. The calculations were performed assuming 7/8-in. diameter tubing because this was the tubing of interest at the time due to the severed tube Content deleted - EPRI Proprietary significant differences are not expected between different tube sizes. Figure 2-13 shows that the crack growth rate increases as the crack grows, eventually achieving a *takeoff* length or angle after which the crack progresses rapidly to tube severance. The time between achieving a given crack length or angle and the time of severance is related to the stress amplitude. Higher stress amplitudes lead to shorter times between a given crack length and tube severance (Figure 2-13).

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Figure 2-13
Fatigue Crack Angles vs. Time: (7/8 Inch Diameter, 50 mil Wall Westinghouse S/G)
(Note That the Initial Angle is Arbitrarily Selected to Allow Display in the Same Figure.)
***Salt* is the Stress Amplitude.**

2.2.2.4 Leak Rate as a Function of Flaw Size for Fatigue Cracks

From calculation results (crack angle versus time [Figure 2-13] and leakage versus time [Figure 2-14]) provided to EPRI, it is possible to derive a fit to the Westinghouse model relating fatigue crack size to leakage. This relationship is well approximated by the following equation (which does not necessarily have a theoretical basis):

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Equation 2-1

where Q is the leakage (in gpd at STP) and x is the crack length (in inches). This relationship is illustrated in Figure 2-15. Also shown in this figure is the corresponding relationship for axial stress corrosion cracks (for nominal inputs, as given in Figure 2-3). Leak rate from fatigue cracks is expected to be much higher than that from ODSCC cracks due to lower tortuosity and roughness associated with fatigue cracks. The Westinghouse fatigue crack model is based on many years of measurements of leakage from fatigue cracks performed by Westinghouse [18]. Additionally, it provides a good approximation of leakage from laboratory fatigue cracks, as presented in the *Integrity Assessment Guidelines* (Figure 9-2 in Reference [1]).

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Figure 2-14
Unsupported U-Bend Fatigue Crack Leak Rate vs. Time: (7/8 Inch Diameter, 50 Mil Wall, Westinghouse SG) Including North Anna Unit 1 Measurements (Leakage in gpd at STP)

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Figure 2-15
Leakage vs Crack Size for Fatigue and Stress Corrosion Cracks (7/8 Inch Tubing)

2.2.2.5 Take Off Leak Rate Increase

Leak rate versus time results from the Westinghouse model are presented in Figure 2-14. Note that the horizontal position of these curves (i.e., the leakage at the initial time) is completely arbitrary. These curves are used only for evaluation of time intervals, so the arbitrary horizontal position is acceptable. The calculation results shown in Figure 2-14 are well fitted by the following equation:

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Equation 2-2

where Q is the leakage (in gpd), Q_0 is the leakage at time zero (arbitrarily set to 15 gpd, which is equivalent to the arbitrary horizontal positioning of the curves discussed above), t is the time (in hours) and a and b are fitting constants. The value of a is the time of rupture. Note that the form of Equation 2-2 and the value of b do not have theoretical significance and are provided for ease of calculation only. For the three curves shown in Figure 2-14, the values of a and b are as follows:

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-

Using this equation, leakage versus time is reproduced in Figure 2-16. Horizontal lines indicating 75 gpd and 150 gpd are shown in Figure 2-16 for reference.

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Figure 2-16
Unsupported U-Bend Fatigue Crack Leak Rate vs. Time: (7/8 Inch Diameter, 50 Mil Wall, Westinghouse SG) (Reproduced)

Using Equation 2-2 the time between reaching certain leakage thresholds and tube rupture can be calculated. Table 2-7 gives examples of these times. For the three stress amplitudes considered here, Figure 2-17 gives the time to reach rupture as a function of the leak rate of change threshold. From Figure 2-17 it is evident that as the rate of change threshold is lowered, the time to rupture is increased, i.e., detecting changes in leakage sooner results in additional time to act before tube rupture.

Table 2-7
Times from Thresholds to Rupture for Fatigue Cracks

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Figure 2-17
Time to Rupture after Reaching a Given Rate of Change of Leak Rate

2.2.2.6 Operating Experience

It is useful to review industry operating experience in order to support the conclusions drawn from the modeling discussed in the previous sections. For this purpose, three primary-to-secondary leak events associated with fatigue cracks are reviewed here. These events are as follows:

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-
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Summary data for each of these are presented in Table 2-8. Additional details are given in the sections below. Overall conclusions regarding industry experience with leakage from fatigue cracks are discussed in Section 2.2.2.6.3.

Table 2-8
Summary Data for Fatigue Crack Leakage Events Reviewed

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2.2.2.6.1 North Anna Unit 1 (1987)

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Figure 2-18
North Anna R9C51 Measured Leak Rate Vs. Time (Arbitrary Initial Time)

2.2.2.6.2 Oconee Unit 2 and Unit 3 (1994)

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Figure 2-19
Oconee Unit 2 1994 Leak Rate (gpd) and Rate of Change in Leak Rate (gpd/hr): Off Gas Monitor

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Figure 2-20
Oconee Unit 3 1994 Leak Rate (gpd) and Rate of Change in Leak Rate (gpd/hr): Off Gas Monitor

From the actual measurements associated with the Oconee leaks it is possible to assess the validity of the calculated results given in Sections 2.2.2.3, 2.2.2.4, and 2.2.2.5. Table 2-9 gives the measured crack angle, calculated crack length (i.e., calculated from the tube diameter and crack angular extent), and leakage as well as the predicted leakage given the crack length and the corresponding crack length given the actual leakage. Content deleted - EPRI Proprietary

Table 2-9
Measured and Calculated Leakage Parameters for Oconee Leakage

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Using Equation 2-2 to fit the data from the Unit 2 leak results in values of *a* and *b* of 36.7 and 23.1, respectively (arbitrarily setting 7/26/94 0:00 as time zero and fitting for the leak rate at time zero). Figure 2-21 shows a selection of leakage data and the fit using Equation 2-2 for the Unit 2 leak. Similarly, the values of *a* and *b* for the Unit 3 leak are 9.34 and 21.7, respectively. Figure 2-22 shows a selection of leakage data and the fit using Equation 2-2 for the Unit 3 leak.

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Figure 2-21
Oconee Unit 2 Leakage versus Time and Fit Using Equation 2-2

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Figure 2-22
Oconee Unit 3 Leakage versus Time and Fit Using Equation 2-2

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2.2.2.6.3 Conclusions from Industry Experience with Fatigue Cracks

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Table 2-10
Calculated Times from Thresholds to Rupture for Operating Experience Fatigue Cracks

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2.2.3 Technical Bases Relative to Other Flaw Types

The models developed in Sections 2.2.1 and 2.2.2 only address leaks from stress corrosion cracks and fatigue cracks, respectively. These flaw types are the focus of these technical bases because they are more likely to lead to a tube rupture and have reasonably predictable growth rates.

However, other flaws can lead to leaks which are less predictable but also less likely to lead to tube rupture. These include the following:

- Foreign Object and Support Structure Wear
- Leaking Tube Plugs
- Wastage
- Pitting
- Manufacturing Defects

Leakage from each of these flaw types is discussed in the sections below.

2.2.3.1 Foreign Object and Support Structure Wear

When a foreign object or support structure such as a tube support plate contacts a tube, flow induced vibrations can cause tube wear at the point of contact. For any given leakage, a wear flaw is expected to have a significantly higher burst pressure than would be expected for a crack. Therefore, leakage limits based on leakage from cracks will be conservative with respect to wear flaws when considering loss of tube integrity. However, such limits may not be conservative when considering the possibility of a sudden increase in leakage from a small, manageable level to a large level with significant operational implications (i.e., a rupture or burst as defined in these *Guidelines*, see Section 2.1). It is possible that a wear patch allowing little or no leakage could fail suddenly leading to a step change in the leakage. As with the extremely rapid fatigue cracks at Mihama and Indian Point discussed in Section 2.1, limits on primary-to-secondary leakage can provide little protection against such events.

A foreign object leak event occurred at Byron Unit 2 in 2002. Byron Unit 2 has Westinghouse D5 steam generators tubed with Alloy 600TT. The tube dimensions are as follows:

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2.2.3.2 Leaking Tube Plugs

One repair method for degraded steam generator tubes is the installation of plugs at the ends of the tubes, isolating them from the primary coolant. Historically, plugs made from Alloy 600 have developed primary water stress corrosion cracking (PWSCC) which has lead to throughwall cracks allowing primary coolant to leak into the degraded tube and then onto the secondary side [20].

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These *Guidelines* do not provide guidance based on this type of leak because industry efforts are considered to have been adequate to address the negative consequences of tube plug leaks. However, for completeness, the following information is included here regarding tube plug leak characteristics which are different from the stress corrosion crack and fatigue crack leaks addressed in Sections 2.2.1 and 2.2.2, respectively:

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-
-

2.2.3.3 Wastage

Early PWRs began operation using sodium phosphate for pH control on the secondary side. For Alloy 600 and Alloy 800NG, this led, in some cases, to wastage on the secondary side [21]. Phosphate usage has been discontinued in PWRs. Therefore, this mechanism is not addressed by these *Guidelines*.

2.2.3.4 Pitting

Historically, steam generator tubing has suffered from pitting corrosion, which in some cases has led to through-wall flaws leading to primary-to-secondary leakage [22]. For a given opening area, stress corrosion cracks lead to a much greater loss in structural integrity than pits. Therefore, leakage limits based on leakage from stress corrosion cracks will be conservative with respect to pits.

2.2.3.5 Manufacturing Defects

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2.2.4 Effects of Secondary Side Deposits and Structures

The correlations between leakage and burst pressure developed in Section 2.2.1 did not include the effects of secondary side deposits or structures. Leakage and burst pressure can be affected by secondary side deposits and support structures. Historically, these issues have been considered in development of these *Guidelines*, although they have not been quantitatively incorporated into the technical bases. They are discussed here for completeness.

Secondary side deposits and structures including the following:

- Tube supports, especially drilled hole tube support plates (which existed in some original steam generators but have largely been eliminated through steam generator replacement)
- Tubesheets
- Packed crevices (such as between tubes and tube supports)
- Sludge piles (especially tubesheet collars – consolidated sludge deposits on the face of the tubesheet which form a collar around the tube)

When considering implementation of leakage limits and interpreting leakage data, the consideration of the following may be useful:

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-
-
-
-
-
-

2.2.5 Other Operating Experience

EPRI maintains a database of primary-to-secondary leakage events as part of the *Steam Generator Degradation Database* [25]. In the preparation of this revision of these *Guidelines*, that database was reviewed [26]. Numerous leakage events have occurred that are not specifically reviewed in these *Guidelines*. The primary reason for their exclusion is that they would not significantly contribute to the technical bases of the *Guidelines*, generally because similar but more bounding events were already included.

Rupture events that were reviewed but not included in the development of the technical bases were the following:

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In the development of these *Guidelines,* the extent to which the risk of tube rupture can be reduced has been weighed against the reasonableness of actions which will reduce that risk. The extent to which leakage limits would need to be reduced in order to have avoided these rupture events was not considered reasonable. Therefore, these events have not formed part of the technical bases of these *Guidelines.*

2.3 Additional Technical Issues

Other technical issues have been considered in the development of these technical bases but have not been specifically incorporated into the specification of operating leakage limits. The consideration of these issues is documented in the subsections below.

2.3.1 Effects of Tubing Alloy

Figure 2-23 shows the following different comparisons of the yield strengths of Alloy 600 and Alloy 690:

- The solid curves show the ASME Code [27] yield strength values for Alloy 600 and Alloy 690 (tube) as a function of temperature. The ASME Code [27] yield strength values for Alloy 600 and Alloy 690 (tube), which are lower limits, are essentially identical when the range for actual yield strengths is considered.
- The vertical lines show the limits given in the EPRI guidelines for Alloy 600 [28] and Alloy 690 [29] tubing procurement, respectively. As can be seen, the range for Alloy 600 includes the range for Alloy 690.
- A number of data for industrial tubing (i.e., tubing which is representative of that installed in operating steam generators) are also shown [30]. These data show significant overlap between Alloy 600 and Alloy 690 (with Alloy 600TT in the same range). For the room temperature values, the averages and standard deviations (calculated using averages for each available heat) are shown in the inset table. Data at higher temperatures are more limited. (Note that the data given in Figure 2-23 are averages for different heats. In general, the distribution within a single heat is much less than the distribution among heats [31].
- Finally, the values for Alloy 600 used in the calculations in Sections 2.2.1 and 2.2.2 (shown in Table 2-1) are also shown (as *Rev 3 values*). These values span a range which includes measurements of both Alloy 600 and Alloy 690 but does not include all of the data for either alloy.

Although installed Alloy 690TT may have a slightly lower operating-temperature yield strength than Alloy 600, the yield strength values for the two materials are within the same range and the difference in changes in yield strength with temperature is small. Therefore, these materials are expected to be mechanically similar. That is, they are likely to have similar structural integrity limiting flaw sizes.

Other material property differences such as ultimate yield strength are also important in the leakage calculations, but quantification of the distributions in other properties are not as available as those for yield strength. However, it is considered likely that within-alloy variation for other material properties are also greater than nominal differences between Alloy 600 and Alloy 690.

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Figure 2-23
Yield Strength of Alloy 600 and Alloy 690 (*Rev 3 Values* Are Those Used in the Burst Calculations in Sections 2.1.1 and 2.1.2)

While Alloy 690 has demonstrated superior corrosion resistance relative to Alloy 600, these *Guidelines* by definition deal with conditions existing after corrosion has degraded the structure of the tubing. Therefore, for the purposes of these *Guidelines*, the corrosion resistance properties of the tubing are not relevant and only the mechanical properties influence the integrity of the tubing.

Based on a review of material specifications and measurements of the mechanical properties of Alloy 600 and Alloy 690 tubing, there appear to be no bases for differences in the treatment of these materials in these *Guidelines*. Specifically, the use of distributions for Alloy 600MA properties in the Monte Carlo analysis described in Section 2.2.1 is adequate to address leakage from Alloy 600TT and Alloy 690TT tubing.

This evaluation has not addressed other steam generator tubing alloys (such as Monel 400 or Alloy 800NG).

2.3.2 Leak Rate Spiking

As illustrated in the figures shown in Section 2.2.1.5, trends of leakage with time generally show non-monotonic behavior in which the leakage rises and falls at different times. The phenomenon of sharp increases followed by a decline is referred to as spiking. As is evident from the operating experience, several different kinds of spiking can occur, including the following:

- The leakage rapidly rises and then steadily decreases to values higher than before the increase. This has been observed with the frequency of spikes on the order of 1 per 5 days (Figure 2-11) and 4 per day (Figure 2-12).
- The leakage increases rapidly and then decreases rapidly to a value essentially the same as before the increase, i.e., spiking starting from and returning to a baseline leakage (Figure 2-9). The frequency of these spikes has been observed in one case to be about 30 per hr or 700 per day.
- The leakage rises and falls in a seemingly random manner while the baseline steadily increases (Figure 2-10). This type of spiking might be characterized as variability in the measurement if the spikes are within the resolution of the monitoring technique.

In many cases, these spikes have been observed using N-16 detectors. Because the half-life of N-16 is approximately 7 seconds, these measurements are essentially instantaneous. Figure 2-9 shows that even for spikes of very short duration (about one minute of elevated leakage and one minute of baseline leakage), the leakage increase is not instantaneous, i.e., values between the baseline and spike leakage are detected.

The causes of various types of spikes are not well understood. However, several reasonable hypotheses have been presented, including the following:

- Ligament tearing: Stress corrosion cracking of Alloy 600 often begins with intergranular attack and proceeds through microcrack development and coalescence into the active crack. This process often leads to multiple cracks in the same general location, especially early in the process when the cracks are small relative to the size of the susceptible area and cracking has not led to a significant re-arrangement of the stress field. The presence of multiple cracks, generally in a parallel pattern as the crack orientation is aligned with the stress field, leads to the creation of ligaments, thin strands of intact metal lying between two cracks. As the crack continues to grow the stresses on the ligament are concentrated until the ligament cracks (this may be enhanced by corrosion processes acting on the ligament itself). This leads to crack growth in spurts, which could be related to spikes in leakage, particularly to the sharp leakage increases observed in some cases.
- Clogging: The decrease in leakage following a spike has been hypothesized to be related to clogging of cracks, either by particulate primary side corrosion particles, by precipitated boron compounds (formed as borated primary coolant flashes to steam as it moves through the crack), or (perhaps only for very thin cracks) oxidation of the crack walls. This would be consistent with the gradual decreases in leakage following a spike. In the absence of some clogging mechanism, the decrease in leakage would imply a reduction in the size of the crack, which does not occur.
- Other causes: In some cases, leakage spikes diminish back to the baseline almost immediately (Figure 2-9). It is difficult to attribute these fluctuations in leakage to ligament tearing or clogging. They occur at a frequency of about 0.03 Hz, which is orders of magnitude below typical thermal-hydraulic stress oscillation frequencies, which are generally in the range of 20-60 Hz [13, 14, 32, 33]. Vibrations on the order of 0.03 Hz are not known to exist in PWRs, but have not been ruled out.

It appears likely that each of these, alone or in combination with other mechanisms, affects the leakage from various flaws leading to the variations in observed spiking phenomena. Other phenomena which could affect leakage with no changes in the crack opening include crack face wear.

The correlations between leakage and flaw size used in Section 2.2.1 have been assessed against operating experience using baseline leakage. Therefore, operating limits based on these correlations (such as those arising from the use of the burst pressure to leakage correlations shown in Figures 2-6, 2-7, and 2-8) are most appropriately based on the baseline leakage, not on the spike leakage. However, for large leaks or rapidly increasing leakage prompt action may be necessary to avoid tube rupture. In these cases, it is appropriate to base responses to leakage on confirmed spikes rather than on baseline leakage values.

2.3.3 Volumetric Versus Mass Leak Rate

The leakage values used in this section to develop technical bases for leakage limits have all been expressed in gallons per day at standard temperature and pressure (at STP), i.e., at room temperature and atmospheric pressure. It is important that plant procedures which govern calculation of leak rate use the same units if leak rate limits developed using these technical bases are used without further modification. Substitution of mass flow rates for the volumetric flow rates used in these *Guidelines* may be made by using the density of water at room temperature. Likewise, conversion to volumetric flow rates at other temperatures may also be made.

2.3.4 Effects of Primary Side pH on Leakage from SCC Cracks

A correlation between primary side pH and leakage has Content deleted - EPRI Proprietary

changes in leakage correlating with changes in pH are independent of flaw size and thus not related to changes in the probability of rupture. However, the current understanding of this phenomenon is not sufficient to warrant its inclusion in the technical bases of these *Guidelines*. The currently available information and its effects on the interpretation of some leakage events are discussed in Appendix A.

2.3.5 Leakage Determination Accuracy

The technical bases discussed in this section and the limits discussed in Section 3 are based on the assumption of accurate determination of the leakage.

One factor that can affect the accuracy of leakage estimates is the primary coolant activity. As illustrated in Equation [5-2], secondary side activity is a function of the product of the leakage and the primary side activity, not the leakage alone. Therefore, any accurate determination of the leakage requires division by the primary coolant activity.

For example, the occurrence of a fuel failure, a breach in the fuel cladding, results in an increase in primary coolant activity. In the case of an increase in primary coolant activity that is not reflected in the leakage calculation (i.e., when an obsolete primary coolant activity is used in the calculation), the calculated leakage will be much higher than the actual leakage. This error would be conservative (leakage would be less than that calculated) and thus might result in action (e.g., power reduction) that would not be required if a more accurate leakage estimate were available. There may be other reasons that result in an increase or decrease or change in the primary side activity.

It is assumed that plant-specific procedures will balance the costs of obtaining more frequent measurements of the primary coolant activity against the costs associated with conservative actions in the absence of more frequent measurements. In developing the limits given in Section 3 from the technical bases presented in this section, it is assumed that plant-specific procedures will require utility personnel to act on a leakage value derived from a specified calculation procedure such that action will not be delayed by further sampling. Section 5 contains detailed discussions related to leakage determination based on primary chemistry activity.

Additional discussion of uncertainty in calculated leak rate is given in References [4] and [5].

2.4 Increased Monitoring Thresholds and Other Industry Guidance

As part of the Revision 4 process, an investigation was made into the historic bases regarding thresholds for increased monitoring. As part of this investigation, other industry guidance documents were reviewed to determine if there were additional technical issues which had not been considered in previous revisions of the *Guidelines*. Section 2.4.1 provides the technical bases for the increased monitoring threshold as given in the initial issue of these *Guidelines* [34]. Sections 2.4.2, 2.4.3, and 2.4.4 summarize guidance provided by the NRC, the Joint Owners Group (which included the Westinghouse and B&W owners groups), and INPO. Section 2.4.5 provides conclusions regarding the relationship between these *Guidelines* and other industry guidance.

2.4.1 Guidelines Threshold of 5 gpd

The initial issue of these *Guidelines* [34] defined normal operation as operation without a detectable leak. It is further stated (Page 2-2):

> Due to the lack of analytical certainty at very low radiochemical concentrations, it is assumed that the leak rate is ≤ 5 gpd.

This language implies that 5 gpd was chosen as the threshold for increased monitoring because it was the lowest leakage that provided reliable quantification. This is supported by the general basis given for both the 5 gpd threshold and the 30 gpd Action Level 1 value in the initial issue of these *Guidelines*, which is as follows (Page 2-5):

> These criteria are consistent with grab sample and RMS sensitivity...

It is presumed that as detection capabilities improved, the language describing 5 gpd as a lower quantification limit was dropped, but the threshold for entering increased monitoring was retained.

2.4.2 NRC Threshold of 3 gpd

Per Reference [35], NRC inspectors are advised as follows (Page 5):

> If steam generator leakage greater than 3 gallons per day was identified during operations or during post-shutdown visual inspections of the tubesheet face, assess whether the licensee has identified a reasonable cause for this leakage based on inspection results.

Inspectors are also instructed to consider several factors in determining whether additional resources should be expended on SG inspection, including the following (Page 7):

> ...a history of primary-to-secondary leakage during the previous operating cycle (e.g., > 3 gallons per day).

For additional technical information about primary-to-secondary leaks, inspectors are referred to Reference [36]. Reference [36] discusses these *Guidelines* and references the 5 gpd limit for increased monitoring established in previous revisions. It also indicates the following:

> It is suggested that the NRC resident inspectors and regional staff use an informal screening criteria of 3 gpd or greater for increased involvement by NRC headquarters staff when steam generator primary-to-secondary leakage is identified. This is not meant to be an absolute threshold, or requirement, because there may be certain instances where there is something unusual about the circumstances of the leakage, or other reason that

> the region would want involvement by the headquarters staff before leakage reaches 3 gpd. If a licensee reports levels of primary-to-secondary leakage exceeding 3 gpd to the resident inspector or regional staff, Office of Nuclear Reactor Regulation (NRR) should be informed through the morning phone calls. The Materials and Chemical Engineering Branch (EMCB) staff is interested in being kept abreast of this information by NRR, Division of Licensing Project Management.

It appears from this language that there is no technical basis for the 3 gpd and that it is based on a NRC desire to be kept informed regarding leakage before the 5 gpd threshold is reached.

2.4.3 TSTF-449, Revision 4

TSTF-449, Revision 4 [37] recommends revision of technical specifications to limit operational leakage to 150 gpd from any one steam generator. This recommendation establishes a uniform technical specification at a much lower limit than the previous limits (720 gpd or 500 gpd through all steam generators). The bases for this limit are the same as those developed in Section 2.2.1, and these *Guidelines* are referenced by TSTF-449.

2.4.4 SOER 93-01

INPO SOER 93-01 [38] was reviewed as part of this effort. It contained no technical considerations for the development of leakage or rate of leakage increase limits which were not already addressed in these *Guidelines*.

2.4.5 Conclusions and Recommendations

All known technical issues discussed in other industry documentation regarding primary-to-secondary leaks are addressed in these *Guidelines*.

2.5 References

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3 OPERATING GUIDELINES FOR PRIMARY-TO-SECONDARY LEAKAGE

3.1 Purpose

The guidance contained in this section is designed so that appropriate actions can be taken early enough to preclude most through-wall tube defects from propagating to rupture as discussed in Section 2. Accordingly, the plant actions become more stringent as leakage or leakage rate of change increases, and immediate power reduction and plant shutdown may be required if leakage trends indicate that a tube defect is rapidly propagating.

This section addresses increased monitoring conditions and Action Levels, and recommends appropriate actions in response to these conditions. The Action Levels contained in this section describe a program that is capable of responding to rapidly propagating steam generator tube defects without relying on time-consuming grab samples. The underlying technical bases for selection of the Action Level criteria are described in Section 2.

3.1.1 General Terminology

The term "activity" as used in these *Guidelines* refers to the concentration of radioactivity in the liquid (μCi/g) or gas (μCi/cc) phases, consistent with standard nuclear power plant practice.

The terms "continuous" and "real-time" have the same meaning.

3.1.2 Leakage

The leakage values used in this section apply to leakage in any one steam generator. Plants with N-16 monitors on individual steam lines may be able to identify the steam generator in which leakage is occurring. If it is not possible to assign the leakage to an individual steam generator, all the leakage is to be conservatively assumed to be from one steam generator.

All actions noted in Section 3 refer to leakage that is calculated in gallons per day (gpd) based on room temperature liquid densities (see Section 2.3.3 for further discussion). For example, 5 gpd equates to ~1.7 lbm/hr, 30 gpd refers to ~10.4 lbm/hr, 75 gpd is ~26 lbm/hr and 150 gpd is ~52 lbm/hr. This is consistent with other similar operational and analytical units and with the technical bases discussed in Section 2.

3.1.3 Failed Fuel

Concerns over failed-fuel maneuverability limits were considered in the formulation of these *Guidelines*. However, imminent tube rupture concerns take precedence over failed-fuel maneuverability limitations.

3.1.4 Frequency

The frequencies for sampling and analysis given in this section are based on the technical discussion given in Section 2. It is understood that in many cases small adjustments to defined intervals may be useful to utilities. For example, a sample that is specified to be taken once every eight hours might be conveniently incorporated into a shift schedule. However, it is likely that, due to variability in the times required for such tasks, such a schedule might result in the interval occasionally exceeding eight hours. The frequency specifications in this section are considered to be sufficiently conservative such that small, reasonable, and non-systematic (e.g., one-time) increases in intervals will not lead to significant increases in risk.

3.2 Definitions

3.2.1 Operating Conditions

Four operating conditions are defined for initiation of station actions based on primary-to-secondary leakage and rate of change of leakage. These are:

- **Modes 3 and 4:** The period of operation during plant heatup or cooldown
- **Mode 1 and 2 Non-Steady State:** The period of operation during reactor startup, shutdown, or low power operations outside the site-specific definition of steady-state operation
- **Steady-State Power Operation:** The Mode 1, steady-state plant condition, as defined in site-specific documents
- **Power Transients:** The period of operation with power transients outside the site-specific definition of steady-state operation that is not associated with startup.

Operating modes are defined by plant Technical Specifications or other regulatory guidance.

3.2.2 Radiation Monitor Status

There are two conditions related to radiation monitors based on the ability to monitor and evaluate primary-to-secondary leakage and action thresholds. These are:

- **Continuous Radiation Monitor:** This condition is when there are one or more radiation monitor(s) available, which meet the following requirements:
  - Continuous operation with an alarm function available in the Control Room, **AND**
  - The monitor is capable of detecting leakage of 30 gpd and higher, **AND**
  - The monitor output is correlated to gpd for continuous monitoring.
- **No Available Continuous Radiation Monitor:** This condition is when there are no continuous radiation monitors.

3.2.3 Monitoring Leakage Conditions and Action Levels

Five conditions, including three Action Levels and one increased monitoring condition, are defined for initiation of station actions based on primary-to-secondary leakage. The conditions are organized in this section starting with the worst case scenario going to the best case scenario, to be consistent with the organization of plant abnormal operating procedures. The conditions are:

- **Action Level 3:** The plant condition, which indicates that the leakage is increasing rapidly and it is mandatory that the unit be promptly shut down to protect the unit from tube rupture
- **Action Level 2:** The plant condition in which leakage has increased to a condition indicating that the underlying flaw has grown to an undesirably large size and it is mandatory that the unit be shut down in a planned manner
- **Action Level 1:** The plant condition in which leakage has increased to a condition that requires frequent monitoring by the radiation monitoring system with periodic benchmarking by laboratory analyses
- **Increased Monitoring:** The condition in which leakage has been detected but is not in a range that can be accurately monitored by most online radiation monitors, does not necessarily indicate imminent risk to steam generator tube integrity, but warrants additional attention
- **Normal Monitoring:** The condition in which detected leakage is less than 5 gpd

Action Levels are intended to be an important line of defense against tube ruptures and against flaws growing to an undesirable size such that the probability of tube rupture is significant under normal or accident conditions. The criteria presented were evaluated against tube ruptures previously experienced by the industry and against tube flaw experience in cases where large flaws were found that had not resulted in rupture. These evaluations suggest that plant shutdown would have been initiated, based on the Action Level recommendations, prior to most tube ruptures, and prior to most flaws growing to a size that would be expected to rupture under accident conditions. However, as discussed in the technical bases section (Section 2), experience shows that, in a small fraction of cases, flaws grow to a large size without detectable leakage and sufficient time to respond.

3.3 Primary-to-Secondary Leak Program Requirements

This section defines the programmatic requirements of a primary-to-secondary leak program. Site procedures or other site documents that contain more restrictive requirements than those contained in these *Guidelines* are to be followed where appropriate.

For power operation, there are two methodologies that can be used to respond to primary-to-secondary leakage:

- **Rate of Change:** Under the rate of change methodology, site-specific procedures and expectations are developed ensuring that the Action Levels in Section 3.5.2.2, which are based on an evaluation of the leakage and the rate of change in leakage, are implemented.
- **Constant Leakage:** Under the constant leakage methodology, site-specific procedures and expectations are developed ensuring that the Action Levels in Section 3.5.2.3, which are based only on leakage rate, are implemented.

A given plant may incorporate either the rate of change methodology (Section 3.5.2.2) or the constant leakage methodology (Section 3.5.2.3) into its primary-to-secondary leak program considering the site specific limitations, procedures, and operating histories. To prevent a plant from changing methodologies during an active leak event, plant program documents are to specify which methodology will be used to respond to primary-to-secondary leakage.

During an active leakage event, power reduction may be initiated in response to Action Level requirements in Section 3.5.2.2 or Section 3.5.2.3. The power reduction may result in entering the "no available continuous radiation monitor" condition. In this situation, plants are to switch from the Action Level requirements in Section 3.5.2.2 or Section 3.5.2.3 and follow the Action Level requirements in Section 3.5.2.1 and the Monitoring requirements in Section 3.4.2.1 or Section 3.4.2.2.

3.3.1 Mandatory Programmatic Requirements

The following are identified as the **mandatory** programmatic elements for the site primary-to-secondary leak program.

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3.4 Monitoring Requirements

Section 3.4 contains the monitoring requirements for operating conditions defined in Section 3.2.1.

3.4.1 Modes 3 and 4

This section applies to Modes 3 and 4 and is associated with startup and shutdown conditions. This section specifically addresses monitoring requirements after an extended period at zero power (e.g., after a refueling outage or extended outages) when the primary coolant has relatively low activity and methods used during steady-state power operation cannot be used. Transient operation not associated with startup and shutdown conditions (power transients) is discussed in Section 3.4.3.

Shall Requirement: Station procedures shall contain a prescribed grab sample or continuous radiation monitor program, as required by Technical Specifications.

Recommendation: The recommended minimum monitoring frequency is once per 24 hours.

3.4.1.1 Plant Conditions

Technical Specifications require monitoring for primary-to-secondary leakage in Modes 1 through 4 during steady state.

Monitoring for primary-to-secondary leakage during startup ensures that the unit does not start up, approaching or exceeding, the primary-to-secondary leakage Technical Specification or other plant administrative limits for primary-to-secondary leakage. Once radiation monitors for primary-to-secondary leakage provide valid trending of leakage values, generally after Mode 1 entry, the monitors are used in conjunction with grab sampling until monitoring requirements for steady-state power operation per Section 3.4.2.2 apply.

Immediately after shutdown the short-lived isotopes are usually at sufficient levels to monitor for leakage using the same techniques that can be used during steady-state power operation as long as the other plant conditions such as operation of the air ejectors and some steam generator steaming allow the measurement. However, the decision on whether all parameters are at a level to make the measurement valid can be time consuming. The use of a tritium calculation for primary-to-secondary leakage may also be considered during this period.

During startups, especially after a long outage like a refueling outage, there are no short-lived isotopes in either the primary or secondary system. Therefore, monitoring for leakage using the same techniques as during steady-state power operation may not be used. This limits measurement of the leakage to chemical or long-lived radiochemical means.

3.4.1.2 Selection of Chemical and Radiochemical Species as a Tracer

Some limitations of using boron, lithium, Co-58, Co-60, Cs-134, and Cs-137 as a tracer are discussed below. A discussion of tritium is also included in this section.

Boron in the primary systems varies depending on where the plant is in the operating cycle. During refueling outages, the boron concentration is usually > 2000 ppm boron, and at these levels, leakage of about 30 gallons per day can be detected by normal analytical methods. Leakage under 30 gallons per day or at lower primary boron concentrations will require a more sensitive analysis. Plants on boric acid treatment, or with significant boron hideout, can receive confusing results in calculating leakage using boron. There is the possibility of boron removal via secondary side anion and/or mixed bed demineralizers, if in service.

Lithium is not usually added to the primary system until late in Mode 4 or early in Mode 3, which limits its use as a tracer on the secondary side. Even after the lithium is added to the primary system, its levels in the secondary system are so low that many analytical methods limit its use.

The use of Co-58 and Co-60 presents problems due to their solubility characteristics in both the primary and secondary systems. Also, for plants with cobalt in the secondary system due to previous primary-to-secondary leakage, the plant will likely see cobalt increases with the startup of each part of the secondary system. Use of cobalt under these conditions will lead to false leakage readings. Even though Cs-134 and Cs-137 do not have the same solubility characteristics as cobalt isotopes, the startup of secondary systems may also identify an increase in the cesium activity as a result of its incorporation in iron oxide deposits. The use of condensate/blowdown demineralizers could provide a large removal source for cesium, making leakage determination more difficult.

Typically the most reliable tracer for primary-to-secondary leakage during early cycle operations is tritium due to the low source levels following refueling outages. Boron, lithium, Co-58, Co-60, Cs-134 and Cs-137 are all possible tracers that could be used during these early periods of operation, but each has limitations. If any of these species are used to detect primary-to-secondary leakage during startup, these limitations are expected to be understood and their use justified.

Tritium is not subject to hideout, demineralizer removal, or solubility concerns. Tritium may be present in the primary system at levels > 0.1 μCi/ml even after refueling outages. Tritium can remain in the secondary systems, but this may be taken into account by quantifying changes in secondary system tritium activity. In multi-unit plants, cross-tying of the units for water conservation can give false indication of primary-to-secondary leakage.

If blowdown is recycled (or steam condensate is used to feed the generators), some tritium will be contained in all steam generators. Sampling the steam generators may yield evidence of elevated tritium activity and help determine which steam generator has the primary-to-secondary leak.

Due to changing plant conditions prior to steady-state operation, calculations for primary-to-secondary leakage, including tritium, are challenging. Therefore, gamma isotopic samples of each steam generator are expected to be analyzed. Should the gamma isotopic analysis indicate that the presence of the principal gamma emitters is greater than the minimum detectable activity level, tritium sampling can be used to verify an actual leak by comparison of each steam generator's tritium activity to the other steam generator(s). A gamma isotopic sample of the steam generators indicating that the principle gamma emitters are less than the minimum detectable limits or at a level that will ensure that calculated off-site dose limits are not challenged during the subsequent startup period may suffice as an alternative to meeting the monitoring requirement of the Technical Specification for primary-to-secondary leakage. Prior to the use of the principle gamma emitter technique to demonstrate compliance with the intent of the Technical Specification leakage limit, the plant is expected to ensure that the Technical Specifications Bases include information allowing the use of this technique.

As startup progresses and the plant is closer to full power operation, the calculations performed with tritium will be more representative of primary-to-secondary leakage. Calculations of primary-to-secondary leakage in non-operational conditions are to be made with the understanding that they are more subject to error than the primary-to-secondary leakage calculations made during operation.

At one plant the lower level of detection using tritium during shutdown conditions was determined to be approximately 50 gallons per day due to the error in quantifying low levels of tritium in the secondary system, which is complicated by sharing of water between units. Measurement errors may be offset by using longer counting times. Analytical sensitivity is expected to allow early detection of a large leak that is near the shutdown or Technical Specification limits.

Sampling of the secondary system and/or steam generators should be performed once every 24 hours. Sampling of primary system tritium may be performed less frequently (i.e., once or twice per week rather than the recommended secondary system sampling frequency of once per 24 hours) since the reactor coolant system tritium activity is not expected to change significantly from day to day during non-operational periods and/or steady-state conditions, unless significant dilution is taking place.

3.4.2 Mode 1 and 2 Non-Steady-State and Steady-State Power Operations

3.4.2.1 Mode 1 and 2 Non-Steady-State Power Operations

This section applies to Mode 1 and 2 non-steady-state power operations.

Content deleted - EPRI Proprietary

Discussion:

This program is expected to take advantage of continuous radiation monitors (defined in Section 3.2.2) to trend primary-to-secondary leakage. The program recognizes the limited ability of the monitors to accurately correlate leakage during changing power levels. When the plant begins to increase power after a long duration outage, isotopes normally used for primary-to-secondary leakage monitoring during power operation begin increasing and any correlation performed prior to the power increase may provide conservative but inaccurate results.

During this time period, the plant will have to rely on a trend of the radiation monitors and grab samples to detect primary-to-secondary leakage. Isotopes will eventually reach a concentration where the radiation monitors can be used as the primary means of detection and grab samples can accurately measure low-level leakage.

For example, at one plant a continuous radiation monitor is correlated to provide a gallon-per-day readout in the control room between 0 and 5% power. During startup, this radiation monitor's alarm is set at 30 gpd, rather than the usual 5 gpd used during normal power operation.

If any alarms are encountered as the power level is increased, power escalation is halted and a grab sample is pulled to verify the leakage and to re-correlate the radiation monitor before power escalation is resumed. Once steady-state power operation is reached, normal primary-to-secondary leakage monitoring practices are used. For plants with N-16 monitors that are automatically correlated to power levels, the plant can use normal primary-to-secondary leakage monitoring practices earlier than reaching steady-state power, depending on the power level at which the monitors are known to provide accurate leakage monitoring.

Section 5 discusses minimization of errors in the leak rate calculations during non-steady-state conditions.

3.4.2.2 Steady-State Power Operations

This section applies to steady-state power operations.

Content deleted - EPRI Proprietary

3.4.3 Power Transients with Known Primary-to-Secondary Leakage

This section applies to transient operation with known primary-to-secondary leakage that is not associated with startup. Transient operation associated with startup is discussed in Section 3.4.1.

Discussion:

During power maneuvers it may be difficult to correlate a continuous radiation monitor to leak rate and the rate of change of the leak rate. Section 3.5.3 provides some recommended actions for this period of operation.

3.5 Action Levels for Various Plant Conditions

3.5.1 Modes 3 and 4

This section applies to Modes 3 and 4.

Content deleted - EPRI Proprietary

3.5.2 Mode 1 and 2 Non-Steady-State and Steady-State Power Operations

This section applies to Mode 1 and 2 non-steady-state and steady-state power operations.

3.5.2.1 No Available Continuous Radiation Monitor

This section applies to Mode 1 and 2 non-steady-state and steady-state power operation when there is no available continuous radiation monitoring as described in Section 3.2.2. Refer to Section 3.4.2.1 for monitoring requirements associated with Mode 1 and 2 non-steady-state and refer to Section 3.4.2.2 for monitoring requirements associated with steady-state power operation.

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Table 3-1
Action When No Available Continuous Radiation Monitor

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Recommendations:

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3.5.2.2 Continuous Radiation Monitor, Rate of Change Methodology

This section specifies actions based on monitoring a combination of leakage and the rate of change in leakage. Section 3.5.2.3 specifies actions based on only monitoring leakage.

3.5.2.2.1 Action Level 3 – Leakage ≥ 75 gpd AND Increasing ≥ 30 gpd/hr OR Leakage ≥ 150 gpd AND Increasing < 30 gpd/hr

Content deleted - EPRI Proprietary

Discussion:

To avoid an unnecessary plant shutdown, Action Level 3 leakage is expected to be *qualitatively confirmed* prior to declaration. Leakage is *qualitatively confirmed* when two independent radiation monitors (typical monitor pairs like off-gas/blowdown monitors, off-gas/N-16 monitors, or N-16/ blowdown monitors) trend in the same direction with the same order of magnitude. Confirmation time is to be kept to a minimum. Precise duplication of leakage measurements, as indicated by the monitors, is not important.

The rate of change limit is provided to identify the potential need for a rapid power reduction, and applies to progressively increasing leakage and not to leakage spikes followed by leakage decreases (see Sections 2.3.2 and 4.1.1.3 for further discussion on leakage spikes).

The 150 gpd limit applies to leakage spikes that are *qualitatively confirmed* (see Sections 2.3.2 and 4.1.1.3 for further discussion on leakage spikes).

Mandatory Requirements:

Content deleted - EPRI Proprietary

Shall Requirement:

Content deleted - EPRI Proprietary

Recommendations:

Content deleted - EPRI Proprietary

3.5.2.2.2 Action Level 2 – Leakage ≥ 75 gpd AND Increasing < 30 gpd/hr

Content deleted - EPRI Proprietary

Discussion:

To avoid an unnecessary plant shutdown, Action Level 2 leakage is expected to be *qualitatively confirmed* prior to declaration. Leakage is *qualitatively confirmed* when two independent radiation monitors (typical monitor pairs like off-gas/steam generator blowdown (blowdown) monitors, off-gas/N-16 monitors, or N-16/ blowdown monitors) trend in the same direction with the same order of magnitude. Confirmation time is to be kept to a minimum. Precise duplication of leakage measurements, as indicated by the monitors, is not important. Plants without a second radiation monitor may use a grab sample to qualitatively confirm the leakage increase, but the Action Level is entered when leakage is 75 gpd or greater and sustained for one hour.

Mandatory Requirements:

Content deleted - EPRI Proprietary

Shall Requirement:

Content deleted - EPRI Proprietary

Recommendations:

Content deleted - EPRI Proprietary

3.5.2.2.3 Action Level 1 – Leakage ≥ 30 gpd

Content deleted - EPRI Proprietary

Discussion:
This condition requires increased attention and monitoring to help ensure the leakage does not propagate rapidly to tube rupture. During this condition, it is important that station personnel have a high degree of confidence in process radiation monitor data so that Action Level 2 and 3 actions will be implemented based on monitor readings rather than grab samples.

Shall Requirements:

Content deleted - EPRI Proprietary

Recommendations:

Content deleted - EPRI Proprietary

3.5.2.2.4 Increased Monitoring – Leakage ≥ 5 gpd

Content deleted - EPRI Proprietary

The presence of detectable primary-to-secondary leakage of 5 gpd or greater suggests that an active degradation phenomenon is occurring. Some phenomena that result in secondary side activity (e.g., a leaking tube plug [see Section 2.2.3.2]) do not propagate rapidly.

Shall Requirement:

Content deleted - EPRI Proprietary

Recommendations:

Content deleted - EPRI Proprietary

3.5.2.3 Continuous Radiation Monitor, Constant Leakage Methodology

This section specifies actions based on monitoring leakage. Section 3.5.2.2 specifies actions based on monitoring a combination of leakage and the rate of change in leakage.

3.5.2.3.1 Action Level 3 – Leakage ≥ 100 gpd

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Discussion:

This entry requirement applies to both sustained leakage and confirmed leakage spikes (Section 2.3.2). To avoid an unnecessary plant shutdown, Action Level 3 leakage is expected to be *qualitatively confirmed* prior to declaration. Leakage is *qualitatively confirmed* when two independent radiation monitors (typical monitor pairs like off-gas/ blowdown monitors, off-gas/N-16 monitors, or N-16/ blowdown monitors) trend in the same direction with the same order of magnitude. Confirmation time should be kept to a minimum. Precise duplication of leakage measurements, as indicated by the monitors, is not important.

The 100 gpd limit applies to leakage spikes that are *qualitatively confirmed* (see Sections 2.3.2 and 4.1.1.3 for further discussion on leakage spikes).

Mandatory Requirements:

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Shall Requirement:

Content deleted - EPRI Proprietary

Recommendations:

Content deleted - EPRI Proprietary

3.5.2.3.2 Action Level 2 – Leakage ≥ 75 gpd

Content deleted - EPRI Proprietary

Discussion:

To avoid an unnecessary plant shutdown, Action Level 2 leakage is expected to be *qualitatively confirmed* prior to declaration. Leakage is *qualitatively confirmed* when two independent radiation monitors (typical monitor pairs like off-gas/steam generator blowdown (blowdown) monitors, off-gas/N-16 monitors, or N-16/ blowdown monitors) trend in the same direction with the same order of magnitude. Confirmation time is to be kept to a minimum. Precise duplication of leakage measurements, as indicated by the monitors, is not important. Plants without a second radiation monitor may use a grab sample to qualitatively confirm the leakage increase, but the Action Level is entered when leakage is 75 gpd or greater and sustained for one hour.

Mandatory Requirement:

Content deleted - EPRI Proprietary

Shall Requirements:

Content deleted - EPRI Proprietary

Recommendations:

Content deleted - EPRI Proprietary

3.5.2.3.3 Action Level 1 – Leakage ≥ 30 gpd

Content deleted - EPRI Proprietary

Discussion: This condition requires increased attention and monitoring to help ensure the leakage does not propagate rapidly to tube rupture. During this condition, it is important that station personnel have a high degree of confidence in process radiation monitor data so that Action Level 2 and 3 actions will be implemented based on monitor readings not grab samples.

Shall Requirements:

Content deleted - EPRI Proprietary

Content deleted - EPRI Proprietary

Recommendations:

Content deleted - EPRI Proprietary

3.5.2.3.4 Increased Monitoring – Leakage ≥ 5 gpd

Content deleted - EPRI Proprietary

The presence of detectable primary-to-secondary leakage of 5 gpd or greater suggests that an active degradation phenomenon is occurring. Some phenomena that result in secondary side activity (e.g., a leaking tube plug [see Section 2.2.3.2]) do not propagate rapidly.

Shall Requirement:

Content deleted - EPRI Proprietary

Recommendations:

Content deleted - EPRI Proprietary

Content deleted - EPRI Proprietary

3.5.3 Power Transients with Known Primary-to-Secondary Leakage

This section applies to transient operation with known primary-to-secondary leakage that is not associated with startup. Transient operation associated with startup is discussed in Section 3.4.1.

Definition: The period of operation with known primary-to-secondary leakage $\geq$ 5 gpd with power transients outside the site-specific definition of steady-state operation.

Recommendations: Since this operating condition implies primary-to-secondary leakage exists, the following actions are recommended for the specified conditions:

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4
CONTINUOUS RADIATION MONITORING

4.1 Monitoring Program and Methods

4.1.1 Elements of an Effective Monitoring Program

As noted in Sections 1.2 and 1.3, an effective primary-to-secondary leakage monitoring program should be capable of handling the following three specific scenarios:

- Low-level and/or slowly increasing primary-to-secondary leakage
- Rapidly increasing primary-to-secondary leakage
- Steam generator tube rupture (no leak before break)

4.1.1.1 Main Characteristics of an Effective System

In order to meet these scenarios, an effective leakage monitoring program cannot rely simply on grab sampling. Rather, as discussed in Section 3, the program is designed such that an operator can respond to rapidly escalating leakage (responses to be initiated within an hour). This can only be accomplished if continuous rapid response data are used to provide a basis for operator actions. Therefore, the leakage monitoring program must use the installed Radiation Monitoring System (RMS) to detect the level and if applicable by site-specific program, the rate of change of radioactivity in the secondary plant. The RMS provides continuous on-line monitoring capability to plant operators for detection of primary-to-secondary leakage. Effective use of RMS alert/alarm setpoints provides early notification of changes in leakage.

Routine plant sampling is also an integral component of the leakage monitoring program. The plant sampling program may be implemented to verify the performance of the RMS, verify alarms, confirm leakage estimates, and provide early detection of levels or changes in radioactivity in the secondary system that are either below the sensitivity of the RMS or are not able to be sensed by the particular type of detector.

As discussed in Sections 3 and 5, it is very difficult to select a single monitored pathway or radionuclide to detect/monitor primary-to-secondary leakage. Both the RMS and sampling programs may employ the concept of "radionuclide diversity." Radionuclide diversity is the idea of using multiple radionuclides to calculate primary-to-secondary leakage. Often these different radionuclides are measured in different locations with different transport times. The importance of comparing multiple primary-to-secondary leakage calculations to ensure the validity of the method used is addressed in NRC Information Notice 94-43 [2]. Not comparing leakage calculated with different methods and attempting to resolve discrepancies have led to non-conservative leakage estimates being used to make operational decisions.

Most plant RMSs include instrumentation for monitoring:

- **Condenser Off Gas**: used to identify the presence of radioactive gases removed from steam condensate
- **Steam Generator Blowdown**: used to identify non-volatile radioactive species in the steam generator bulk water (excluding OTSGs)
- **Main Steam**: used to detect volatile gases, and in some cases N-16, carried from the steam generator via the main steam

To identify the presence of primary-to-secondary leakage and to permit its calculation, routine grab samples may be collected from:

- **Reactor Coolant**: used to quantify the source term
- **Steam Generator Blowdown**: used to detect non-volatile radioactive species in liquid
- **Condenser Off Gas**: used to detect noble gas and other volatile species removed from steam condensate
- **Condensed Main Steam**: used to detect noble gas and other volatile species carried over with main steam
- **Condensate**: used to detect tritium, iodine and other soluble species
- **Blowdown Filters and Ion Exchanger Columns (or ion exchange materials used for chemistry sampling that receive continuous flow):** used to detect particulates and ionic species from liquid streams

4.1.1.2 Trending Leakage via the RMS

As noted in Section 4.1.1.1, plant designs typically include radiation monitors on the condenser off gas, main steam, and steam generator blowdown (for RSGs). A radiation monitor is better suited for trending leakage than grab sample analyses, especially if the monitor is trended on a data acquisition system. Some of the advantages are as follows:

- Low manpower intensity - unlike consecutive grab sampling
- No time delay from sample to results, i.e., the reading is instantaneous
- No gaps in the data points, i.e., the reading is continuous
- Not subject to the variations associated with the parameters of grab sampling

Typical radiation monitoring systems may not provide results as accurate as those of a grab sample procedure. However, the advantages given above are thought to be much more important than this lack of accuracy.

Another major drawback to use of the RMS is that radiation monitors read out in dose, counts, etc., not leakage. This drawback can be overcome by correlating the monitor reading using isotope sensitivity either from the manufacturer's calibration curve or an in-house generated curve. Appendix C discusses calculations supporting the development of such a correlation. This correlation can be used to estimate leakage from the monitor reading until this estimated leakage can be correlated with an actual leak.

The best RMS correlation would be based on data from at least three leakage events that span the workable range of the monitor, including a leak of large magnitude. However, in reality plants are not likely to have such data. Instead, plants have typically correlated monitors using data from small leaks that are more likely to occur. This correlation process may be accomplished by determining the leakage as accurately as possible with several methods, including tritium. Then, a correlation of leakage with the reading of the particular monitor may be determined for trending purposes.

For example, leakage was determined to be 5 gpd by the tritium method and the condenser off gas monitor was reading 1000 CPM. These values correlate to a monitor response of 200 CPM/gpd. If the count rate increased to 1500 CPM, then the leakage could be assumed to have increased by 2.5 gpd to a value of 7.5 gpd. A single point calibration may not be linear over the range of the instrument, but it could be used to determine an action level for grab sample analysis to obtain an accurate leakage or used in tracking the increase of rapidly propagating leakage. Appendix C provides an in depth example of using an air ejector monitor for determining leakage.

4.1.1.3 Instrument Spikes

Radiation monitors occasionally exhibit spikes as a result of electronic noise. Spikes due to electronic noise can generally be identified and distinguished from real spikes in primary-to-secondary leakage by checking the response of radiation monitoring pairs. For example, a blowdown radiation monitor may be used to check whether a spike in the off gas monitors was a real spike in activity or was due to electronic noise. For this reason, it is desirable to have two or more operational continuous radiation monitoring systems.

4.2 Evaluation of Monitoring Methods

Detection capability and measurement uncertainty are key elements for consideration when selecting which monitoring method to use and/or when evaluating the best way to apply a given method.

4.2.1 Factors Affecting Leakage Detection and Measurements

Detection capabilities and measurement uncertainties are dynamic, as opposed to fixed, parameters. The specific values may vary with plant operating status and/or history. Detection capability and measurement uncertainties are a function of the following parameters:

- **Source Term**: Source term is the activity that exists in the primary system. Larger source terms enhance the leakage detection capability and lower the uncertainty due to improved counting statistics. Plant operating status and history can have a significant impact on the source term. The production of activation products (e.g., N-16, Ar-41, Na-24, etc.) depends on reactor power level and, with the exception of N-16, the RCS concentration of the material being activated. Production of fission products (e.g., Xe-133, Xe-135, Kr-88, I-131, etc.) depends on core operating history and fuel integrity. Tritium production is influenced by boron concentration and by water management practices (e.g., RCS evaporator distillate recycle/discharge). Finally, operations such as RCS degassing or dilution at the end of cycle can also influence the RCS source term. Plants that have very low noble gas activity in the reactor coolant (due to deaerated makeup or multiple cycles without a fuel leak) may consider adding Ar-40 to the RCS to increase the RCS Ar-41 concentration (Appendix E). Appendix D provides additional information about quantifying leaks during non-operating conditions, when source terms may be particularly low.

- **Primary-to-Secondary Leakage**: The leakage determines the rate at which activity is released into the secondary system. For a given source term, higher leakage leads to higher activity in the secondary side and lower relative uncertainty due to improved counting statistics.
- **Sample Transport Time**: The sample transport time includes time for mixing as well as transport to the radiation monitor. The location of the leakage can impact transport time as well as mixing within the steam generator. Active leakage in a free span region will provide significantly different data than leakage from a region with less communication with the steam generator water (e.g., a leaking tube plug or deep tubesheet crevice leakage). Transport time becomes very important for radionuclides with short half-lives and when comparing readings from different secondary system sources.
- **Properties of the Radionuclide Measured**: The properties of the isotope being measured by the detection system that affect its sensitivity include solubility in water (partition coefficients), chemical interactions (plate out), hideout, half-life (decay), parent/daughter ingrowth (species from transformation), and decay scheme (type of radiation emitted).
- **Detector Efficiency**: Detector efficiency refers to the response of the detector used to measure a particular radionuclide as a function of the type and energy of the radiation measured. In the case of gross channel analyzers, such as those commonly found in plant RMS monitors, the systematic errors associated with the monitor readings caused by the specific radionuclide energy response can be significant unless a correction is made for the specific isotopic mix.
- **Detection Sensitivity:** Detection sensitivity is the ability of the detection system to distinguish between signal and noise response. All monitors and laboratory instrumentation have a lower limit of detection (LLD) based on the system design parameters and the type of detector. Sensitivity can be enhanced by ensuring the sample is not diluted by other liquid/gas streams.

The parameters listed above are interrelated and dependent on the expected operating conditions.

4.2.2 Detection Capability

The detection capabilities of the instrumentation and sampling program are to be consistent with the operational responses presented in Section 3. The detection capability of the primary-to-secondary leakage measurements may be sufficient to identify leakage which could potentially cause secondary side radiological problems and/or are rapidly increasing (e.g., due to very rapid tube degradation).

Under certain plant operating conditions (e.g., startup, shutdown, etc.), the detection capability of the instrumentation may not be sufficient to provide indication of leakage or changes in leakage at the established action levels described in Section 3. Under these conditions, it may be necessary to implement frequent grab sampling in order to attain the required sensitivities.

Determination of the detection capabilities would typically be performed at least once per fuel cycle and any time there is a change to the input parameters that could affect the detection capabilities in a negative manner and challenge the detection capabilities.

Detection capabilities are typically assessed for the instrumentation and sampling techniques used to detect leakage. The different techniques use the calculational methodologies developed in Section 5. The minimum leakage detection capability can be calculated by substituting the following into the appropriate equation described in Section 5:

- **For Sample or Radiation Monitor Concentration**: Substitute the minimum detection capability (LLD) of the radiation monitor or grab sampling technique as the measured concentration. For radiation monitors that operate in a gross counting mode and respond differently depending on the specific isotopic mix of the process stream monitored, the calculation becomes more complicated. In these cases, the monitor response can be related to the isotopic mixture of the RCS source term using energy response data obtained during the primary calibration of the detector. In addition, the lower limit of detection for the monitor is influenced by monitor background. In locations where monitor background is elevated, reevaluation of the detection sensitivity of the monitor may be desirable. Actions can be taken to reduce elevated instrument background that is adversely affecting leakage detection sensitivity.
- **For RCS Concentration**: Substitute the current or most recent RCS source term or, if unavailable, substitute values that are based on an assumed (from UFSAR and/or historical data) typical concentration.
- **For Other Operational Parameters (system flow rates, etc.):** Substitute actual measured parameters or, if unavailable, substitute typical (based on design or historical data) parameter values.

4.2.3 Measurement Uncertainty

Systematic (or bias) and random errors introduce uncertainty into the values of measured parameters and influence the accuracy and precision of these measurements. The uncertainties in the measured values used to calculate the values of other parameters introduce uncertainty into the values of these calculated quantities (measurement uncertainty and error propagation are discussed in detail in Reference [7], which was completed in support of these *Guidelines*). These errors influence the accuracy and precision of the measurement. An analysis of the methods used to quantify leakage should be performed.

If a measurement is accurate, then it has a low degree of systematic error. A measurement is accurate if it is very nearly the actual value of the parameter being measured. The accuracy of a measurement is important when the measurement value (or calculated leakage) is compared with program action levels. As discussed in earlier sections, sample hideout and decay, or some other removal mechanism (e.g., chemical reaction, ion exchange, etc.), can induce a systematic error into the measured result and ultimately into the calculated leakage.

A simple method to test the accuracy of the result is to compare leakage calculated using actual grab sample data for different isotopes and sample media considering the Section 4.1.1.1 discussion on radionuclide diversity. If there are large but reproducible discrepancies in the calculated leakage, one of the measured results most likely contains a systematic error.

Therefore, it is important to compare leakage calculated by different methods in order to cross validate the accuracy of the calculation. If the systematic error can be determined, a correction can be applied to the inaccurate measurement.

If a measurement is precise, then it has a low degree of random error. Precision quantifies the reproducibility of a measurement and should not be confused with accuracy. A precise measurement can be inaccurate if the measurement contains a systematic error. Under these circumstances, precise but inaccurate measurements can provide reliable trend information. In fact, the more precise the measurement - the more sensitive the parameter may be for identifying trend changes. A good example would be monitoring I-131 levels in steam generator blowdown even with a significant amount of hideout. Although using the measured iodine concentration to calculate leakage underestimates leakage, small changes in the relative leakage are easily detected because of the high precision of the laboratory analysis method.

4.2.4 Classification of Monitoring Methods

Monitoring methods can be classified as either qualitative or quantitative depending on how the method is applied. The classification of each method depends on the uncertainty associated with the monitored parameter or estimated leakage. Both methods are useful in monitoring and assessing changes in primary-to-secondary leakage, but it is important that personnel performing leakage calculations have guidance on when to utilize each method as well as the limitations associated with each method.

Quantitative methods generally have low uncertainties (both accurate and precise) and provide a reliable estimate of the actual parameter measured. Quantitative methods are used to determine leakage monitoring program action levels. Currently accepted quantitative methods include leakage calculated using noble gas activity in condenser off gas samples and, for low-level leakage, tritium in secondary system samples (provided that corrections can be made to account for any other sources of secondary side system tritium contamination such as cross-unit contamination resulting from shared auxiliary steam supply). With respect to the RMS, reasonable estimates of primary-to-secondary leakage can be made using condenser off gas monitor readings (provided that the monitor response has been corrected to either the RCS source term isotopic mixture or correlated to actual grab sample data) and main steam N-16 monitors (provided the system has been calibrated for geometry, detector efficiency, RCS activity, transport time, etc). Other methods may fall into this category provided that sufficient operational experience is available to confirm the validity of the method by comparing results derived using that method with those obtained using other independent quantitative methods. Methods that are subject to a fixed systematic error can also be used to determine leakage provided that the measurement result is corrected for the inherent systematic error.

Qualitative methods can be either imprecise but accurate or inaccurate but precise. An example of a measurement that is inaccurate but precise would be using steam generator blowdown measurements of I-131 and Na-24 to trend leakage changes in the presence of active hideout and decay. An example of an accurate but imprecise measurement would be calculating leakage using a radionuclide that has low systematic error, but that has large measurement uncertainty due to a low concentration (poor counting statistics). It should be noted that most leakage calculation methods fall into this category.

4.3 Radiation Monitoring Programs

Section 4.3 is a compilation of information related to radiation monitors for site staff and is for information only.

4.3.1 Leak Detection and Monitoring Instrumentation

Radiation monitors available in most plants are located in steam generator blowdown, condenser off-gas, and main steam lines. In addition to these, some facilities have also installed N-16 monitors that supplement the main steam line monitors required by Regulatory Guide 1.97 [5]. Finally, some facilities utilize portable instrumentation and area monitors to supplement the existing installed instrumentation. The types of portable instrumentation available include N-16 or other portable radiation monitors and survey instruments. Each type of monitor is discussed in greater detail in the following sections.

4.3.1.1 Steam Generator Blowdown Radiation Monitors

The blowdown monitors typically used are liquid monitors in an off-line sampling configuration. A sodium iodide detector is the most common type used. The monitor is operated in a gross counting mode. The monitors detect soluble gamma emitters in steam generator blowdown. Since they are operated in a gross counting mode, monitor response depends on the radionuclide mix of the sampled stream. Typical lower limits of detection range from approximately 1E-7 to 1E-6 μCi/g for effluent and process monitoring applications, respectively.

The monitors are typically relied on to provide qualitative information on primary-to-secondary leakage. Due to their sensitivity, they are responsive to small changes in activity. When no radioactivity is present or at concentrations less than the sensitivity of the monitor, setpoints are typically set at some multiple of background that prevents spurious alarms but still provides early warning of increasing radioactivity. Once detectable activity is present, leak rates calculated based on a quantitative grab sample method can be correlated directly to monitor readings. In this instance, the assumption would be made that if the leak rate doubles, the monitor reading would be expected to double. This assumption is subject to significant errors as discussed below. Caution is warranted with respect to acting solely on the blowdown monitor readings without confirmation from another monitor or grab sample. Operational actions, such as a power decrease, can initiate hideout return and cause steam generator activity to increase without a corresponding increase in leak rate.

Limitations associated with using these monitors for performing quantitative leak rate assessment include:

- Hideout: If radionuclides are deposited in steam generator crevices or hide out within oxides and sludge prior to reaching the monitor, the blowdown leak rate calculation will underestimate the actual leak rate. As noted above, hideout return can lead to a false indication of increasing leakage during power reduction.
- Dependency on Radionuclide Mix: Due to operation in gross count mode, the monitor response is dependent on the specific radionuclide mix sampled. If the radionuclide mix changes, errors in monitor readings can result. For example, during a reactor downpower, if a significant amount of hideout return occurs, the radionuclide mix of the sampled stream will

be altered. In addition, erroneously high readings will be indicated with no change in leak rate. Therefore, alarms on these monitors need to be verified using an independent pathway (typically main steam N-16 monitors or the condenser off-gas monitor).

- Response Times: Another problem that has occurred with these monitors is slow response times due to relatively low sample flows through long sample lines. In fact, if a steam generator tube rupture were to occur, there is a possibility that containment isolation could be actuated by a safety system prior to the blowdown sample reaching the monitor. If containment isolation isolates the monitor sample lines from containment, the monitor may never respond to the steam generator tube rupture (SGTR) event. It is important that monitor response time be examined. Response time can sometimes be decreased without making design modifications by increasing sample flow.
- Background Radiation Levels: If high background contamination levels exist in the secondary system (due to a prior leak or a large active leak), the sensitivity of the monitor to detect changes in activity will be reduced due to higher background readings.
- Fuel Failures: In the event of a fuel leak or an increase in a fuel leak in the presence of active primary-to-secondary leakage, the monitors will reflect higher primary-to-secondary leak rate. Depending on the level of the fuel failure, this may drive the primary-to-secondary leak rate as indicated by the monitor into higher action levels. Therefore, it is important to ensure the change in the indicated primary-to-secondary leak rate is correlated against the failed fuel radiation monitor (if available) or new reactor coolant noble gas samples.

4.3.1.2 Condenser Air Removal Radiation Monitors

The typical condenser air removal radiation monitors monitor non-condensable gases discharged from the condenser. The monitors are sensitive to gaseous activity. There are several sampling configurations available: off-line, in-line (or in-duct), and adjacent-to-line. The detectors commonly used are Geiger-Muller (GM) detectors or organic (beta) scintillation detectors, operated in a gross counting mode. The organic scintillation detectors respond to beta emission from the gaseous activity discharged from the condenser. Since these monitors are operated in a gross counting mode, their responses depend on the radionuclide mix of the sampled stream. GM detectors are sensitive to both the beta and gamma emissions (if used in an in-line or off-line configuration) and gamma emissions (if used in an adjacent-to-line configuration) from radioactive gases in the condenser off-gas. The energy response of the GM tubes depends on the window thickness and material used. In some applications, the monitors may perform a dual role. The monitors may function as both a process monitor and effluent radiation monitor. Typical lower limits of detection range from approximately 1E-7 to 1E-5 μCi/cc for in-line/off-line monitors and adjacent-to-line monitors, respectively.

Like other monitors, there are limitations associated with using condenser air removal monitors for evaluating primary-to-secondary leakage. These limitations include:

- The energy response characteristics of the associated detectors operating in a gross counting mode may affect the accuracy of calculated leakage and should be considered. In order to provide accurate readings, monitor response should be corrected for the specific isotopic mix of the sample stream. The radionuclide energy response can either be calculated or, if activity is present, directly correlated to monitor readings by comparing grab sample data to monitor readings obtained during sampling.

- Accurate measurement of process flow may also affect the accuracy of calculated leakage. In order to calculate leak rate, the process flow past the monitor must be known. Because the process stream consists of a moisture saturated vapor that can contain water droplets, the effect of this sample steam needs to be considered on the instrumentation used to measure process flow. In addition, for utilities that do not have process flow instrumentation, the accuracy of the leakage estimates would be limited by the accuracy of the assumed process flow. Therefore, estimates of process flow should be confirmed by measurement or ensured to be conservative.

Limitations notwithstanding, the condenser air removal monitors provide the most accurate estimate of primary-to-secondary leakage for many leak scenarios. Readings from these monitors can be used to give a rapid assessment of leak rate to operators. Although these monitors provide reliable leak rate information, when looking for rapid increases in leak rate, the sample transport time from the condenser to the monitor should be evaluated. Typically, those condenser systems that operate with vacuum pumps have low flow rates (on the order of a few cubic feet per minute) and large diameter exhaust lines (to accommodate the high flow rates typically encountered when initially pulling vacuum). If the monitor is not located near the condenser, transport time could be significant. Although the monitor would respond to leakage increases, the response might not be observed until some time after the event occurs, e.g., 5 or more minutes.

If no activity is present in the process stream or if it is less than the sensitivity of the monitor, alert/alarm setpoints for these monitors should be set as low as possible without causing spurious alarms in order to provide early indication of primary-to-secondary leakage. If the energy response characteristics of the monitor are known (usually this information is available from primary calibration data), the setpoint can be set to correspond to a leak rate action level by using actual RCS activity. If the monitor is used in effluent applications, the setpoints used to relate monitor readings to off-site dose may not provide early alarm indication of changing leak rate. Refer to Appendix B for corrections that should be considered for leak rate calculations associated with condenser off-gas.

4.3.1.3 Main Steam Line Monitors

4.3.1.3.1 N-16 Main Steam Line Monitors

These monitors are typically mounted on the main steam lines in an adjacent-to-line configuration. A large volume sodium iodide detector is typically used. The monitor is usually operated in either a multi-channel analyzer mode with an energy window set to detect the N-16 6.13 MeV photopeak or in a gross counting mode with a lower level discriminator set to detect only high-energy gamma radiation. The effectiveness of leakage monitoring via N-16 detectors can vary depending on the leak scenario. For example, it has been calculated that the N-16 concentration resulting from an 11 gpd leak can range from 2.4E-5 μCi/cc to 4.0E-14 μCi/cc, depending on whether the leak is from an unplugged, leaking tube or from a flawed tube with a leaking plug. However, these monitors can be very effective for most leak scenarios that will lead to rupture. Because N-16 has a 7 second half-life, sample transport time to the monitor becomes significant. In most calculations, the sample holdup time in the steam generator becomes the limiting factor. Small errors in the estimate of the hold up times in the generator can result in significant errors when calculating leakage based on N-16 monitor response. Therefore, N-16 monitor response to leakage is normally determined empirically by correlating indicated monitor response to known leakage calculated using grab sampling. This correlation is only specific to the particular leak and may not be transferable to leaks occurring at different locations in the generator.

Some of the advantages and challenges of N-16 monitors are:

- Depending on monitor design, N-16 monitors may be useful at less than 40% power. In most cases the validity of power correction is bounded on the lower end at 20% power. Primary N-16 activity concentration decreases approximately linearly with power. Steam generator and main steam line mass transport times increase significantly as power decreases. Therefore, monitor response can be significantly modified due to N-16 decay and the use of N-16 monitors at low power should be evaluated and its limitations understood by plant personnel. Depending on monitor design, N-16 monitors can be used below 20% load by using local electronic meters that read out in CPM.
- Due to the high flow velocity in the steam lines, the transport time from the generator to the radiation monitor is typically very short. Thus, the monitors respond almost instantaneously to increases in leakage. However, the half-life of N-16 is short and there can be considerable error associated with estimating leakage, unless the monitor readings are correlated to leakage calculated using grab sampling.
- An N-16 monitor can also provide diagnostic information. For example, if grab sampling at the condenser exhaust indicates significant leakage, but there is no N-16 activity detected, the leak may be the result of a leaking tube plug or sleeve or by a crack in a deep tubesheet crevice (i.e., a leak scenario with significant holdup time in the steam generator). This observation could also indicate that the source of the activity in the condenser exhaust is from a source other than primary-to-secondary leakage.

Because N-16 has a 7 second half-life and there are no other isotopes with a photopeak energy near that of N-16, response checking and calibration of N-16 monitors can be performed with a special source or other qualified process depending on vendor recommendations. A reference source consisting of Cm-244/C-13 can be used to generate an excited state of O-16 by an (α,n) reaction with the C-13. As the excited state of O-16 decays, a 6.13 MeV gamma is emitted.

Another difficulty involves using a sodium iodide detector with a fixed window. N-16 monitors have built-in sources that compensate for temperature changes and detector drift, which minimizes this problem.

Detector "cross talk" can cause problems related to using N-16 monitors. Cross talk is simply the response of one steam line monitor detecting high-energy gamma radiation emitted from an adjacent steam line. For steam lines that are in close proximity to one another, a significant response (on the order of 10 to 25% for two foot diameter steam lines that are seven feet apart) can be observed on a steam line monitor that may be monitoring an unaffected generator. This can present difficulties to operators when attempting to assess an affected generator or if tracking leaks that occur in more than one generator. Cross talk may also be used to confirm the adjacent N-16 monitor's indication. Cross talk may be minimized by relocating portable N-16 preferentially across the steam line in question to yield better results.

Once monitor readings are correlated with leakage calculated using grab sample data, the monitor readings can provide a direct measurement of leakage. An increase in monitor readings would suggest that the leakage increased by the same factor. However, it is possible for the leakage to increase without any response on the N-16 monitor. For example, if a new leak develops that has a significantly different transport time than the leak causing the initial response. Therefore, N-16 leakage estimates should be periodically compared to leakage calculated using chemistry grab sample data to ensure that the correlation remains valid.

With no detectable activity, alert/alarm setpoints should be set as low as possible to alert operators to potential changes in leakage while minimizing spurious alarms. If detectable activity is present in the steam line, the monitor reading can be correlated to the calculated leakage based on grab sample analysis. The alarm setpoint on the monitor can then be set to correspond to a leakage action level.

4.3.1.3.1 Non N-16 Main Steam Line Monitors

The main steam line monitors discussed in this section are installed at most facilities as required by Regulatory Guide 1.97 [5] and not intended to be used for low-level primary-to-secondary leakage, but relied on for design bases accidents. Due to the high pressure and temperature of the process stream, these monitors are typically installed in an adjacent-to-line configuration. The detectors used are either ion chambers, GM tubes, or in some applications, sodium iodide detectors. The monitors respond to the gamma rays emitted from the radioactive gases and vapors being carried through the steam lines. The accuracy and range requirements for these monitors are specified by Regulatory Guide 1.97 [5]. and ion chamber monitors typically read out in gamma dose rates. Calculations are necessary to estimate the actual activity in the main steam lines. Typical ranges are 1E-4 R/hr to 1E+2 R/hr (corresponding to fission product concentrations between 1E-3 to 1E+3 μCi/cc after applying calculated conversion factors).

The major limitation of these monitors is that they are not sensitive to small changes in leak rate. Because these monitors measure activity through a steel pipe that is about an inch thick, the sensitivity to isotopes with low-energy gamma rays (such as Xe-133 with an 80 keV photon) is minimal. These monitors would only respond if there were sufficiently high RCS source terms. As a result, these monitors cannot be used for low-level leakage detection and are limited to post-accident assessment of significant releases. However, these monitors will exhibit a response to N-16, but typically only at levels corresponding to leakage in the gallons per minute range. Because of the N-16 response, these monitors can provide a clear indication of a SGTR *if* the rupture occurs while the unit is at power. However, once the reactor trips the readings will typically return to background levels.

Because of the low sensitivity of these monitors under normal failed fuel conditions and low-level leak rate, they typically do not provide useful trend information. Alarm setpoints are typically set at three times background. In most facilities with such monitors, the setpoint is determined by the plant Technical Specifications.

4.3.1.4 Portable Instrumentation

Some utilities use a portable system to detect N-16 in the steam lines. The system consists of a portable multiple-channel analyzer (MCA)/amplifier/power supply coupled to a sodium iodide detector. The instrumentation can be used as a diagnostic tool to identify the affected generator and possibly obtain information concerning the cause of the leak. Refer to the previous section for more detailed information regarding N-16 monitoring.

A few utilities use area monitors attached to the side of piping, ion exchangers, or flash tanks to provide a qualitative indication of changing leak rate. In addition, some facilities have incorporated into their station operating procedures instructions on how to perform surveys on ion exchange columns on in-line analyzers in the laboratory to provide a rapid method for confirming increasing radioactive contamination or assessing the affected steam generator.

The information obtained from portable survey instrumentation and/or area monitors can provide a qualitative indication of changing leak rate. Readings should be correlated by observing readings on another monitored pathway. Finally, if significant activity is present in the secondary system from either an earlier tube leak or an active leak, the information obtained may be masked because of the reduced sensitivity of the monitoring instrumentation caused by high background.

4.4 Radiation Monitoring Systems for Primary-to-Secondary Leakage Monitoring Programs

As discussed in several locations within this section there are several good practices that if implemented, are expected to enhance the capabilities of radiation monitoring systems. Table 4-1 provides a collection of these practices for consideration.

Table 4-1
Radiation Monitoring System Practices

Area	Practice	Section
Trending	The best RMS correlation would include at least three points that span the workable range of the monitor including one that corresponds to a leak of large magnitude.	4.1.1.2
Trending	Typically the monitor is correlated using small leaks that are more likely to occur using as many methods as possible to measure leakage as accurately as possible for the adjustment of the monitor correlations.	4.1.1.2
Instrument Spikes	It is desirable to have two or more operational continuous radiation monitoring systems. This contingency provides a backup monitor in the event of spiking.	4.1.1.3
Source Term	Plants that have very low noble gas activity in the reactor coolant may consider adding Ar-40 to the RCS to increase the RCS Ar-41 concentration. (See Appendix E.).	4.2.1
Detection Capability	Determination of the RMS detection capabilities should typically be performed at least once per fuel cycle.	4.2.2
Detection Capability	Determination of the RMS detection capabilities should be verified any time there is a change to the input parameters that could negatively affect the detection capabilities and challenge the detection capabilities.	4.2.2
Detection Capability	Plant personnel should trend detector background readings and reevaluate as required by plant procedures to ensure that background noise levels do not adversely impact detection sensitivity.	4.2.2

4.5 References

1. USNRC Information Notice No. 91-43: "Recent Incidents Involving Rapid Increases in Primary-to-Secondary Leak Rate," (July 1991).
2. USNRC Information Notice No. 94-43: "Determination of Primary-to-Secondary Steam Generator Leak Rate," (June 1994).
3. Code of Federal Regulations, Title 10 Part 50, "Domestic Licensing of Production and Utilization Facilities."
4. USNRC Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," (May 1973).

5. USNRC Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," (December 1980).
6. *Calibration of Radiation Monitors at Nuclear Power Plants.* EPRI, Palo Alto, CA: 2005. 1011965.
7. M. Dumouchel (DEI), memo to D. Perkins (EPRI), August 31, 2010. M-5593-00-08, Rev. 2.

5 LEAK RATE CALCULATIONS

5.1 Introduction

The recommended operational responses to primary-to-secondary leakage discussed in Section 3 rely on accurate assessments of the leakage. This section identifies how to calculate leakage based on isotopic analyses of various secondary system samples. The information contained in this section quantifies some of the mathematical "unknowns" required for leakage calculations. These can be used if plant-specific values have not yet been determined.

The calculations provided in this section are based on specific assumptions. All actions noted in Section 3 refer to leak rates that are calculated in gallons per day (gpd) at room temperature (see Section 2.3.3 for further discussion). Care should be exercised by each utility to ensure the plant-specific conditions are bounded by these assumptions, or appropriate modifications should be made to the calculations as necessary. These equations assume that A_{RCS} is reactor coolant activity corrected for sample delays. Some utilities may be able to demonstrate under some condition-specific circumstances transport times are significantly shorter than decay times and corrections may not be required for some of the longer lived nuclides. Site specific evaluations should be performed before making this assumption.

5.2 Leak Rate Calculations via Condenser Off Gas Analysis

5.2.1 Introduction

Dissolved radiogases in the RCS pass into the secondary side of a steam generator when a primary-to-secondary leak exists. These radiogases are quickly transported out of the steam generators with the main steam and are removed from the condensing steam by the condenser air removal system. Quantification of the primary-to-secondary leakage can be made by comparing the radiogas activity removed through the condenser off gas system (neglecting the solubility of the radiogases in condensate) to the radiogas in the reactor coolant. This method of leakage quantification provides a total primary-to-secondary leakage and does not identify the leaking steam generator(s).

The condenser off gas analysis for leakage determination has several major advantages over the other methods that make it the preferred method under most (but not all) conditions. Some advantages of this method are as follows:

- It is universally applicable to both RSGs and OTSGs.
- It utilizes noble gas isotopes which makes it unnecessary to factor partition effects or other chemical/physical reactions.
- This method provides a rapid leak rate determination because equilibrium considerations are unnecessary (i.e., it is assumed that the noble gases are 100% removed by the condenser air removal system).

5.2.2 The Basic Relationship

The basic relationship for leak rate measurement using condenser off gas analysis is given as follows:

$$LR = \frac{A_g F_g C}{A_{RCS}}$$ **Equation 5-1**

Where:

LR	=	Primary-to-secondary leakage (gpd)
A_g	=	Activity concentration of noble gas radionuclide in the condenser off gas sample (μCi/cc)
A_{RCS}	=	Activity concentration of noble gas isotope in the reactor coolant (μCi/g)
F_g	=	Flow rate of the condenser off gas (scfm)
C	=	1.08×10^4 (gal•cc•min)/(g•ft^3•day), conversion constant which includes: 60 minutes per hour 24 hours per day 28317 cc/ft^3 1 gal per 3785 ml 1 ml per g reactor coolant

This calculation can be used to determine instantaneous leak rate in accordance with following assumptions:

- No significant condenser off gas sample transport decay effects (see Section 5.2.5 and Appendix B for additional discussion)
- No significant mother/daughter decay relationship effects
- RCS noble gas concentrations remain constant, i.e., no power transients, RCS degassing, etc.
- All of the noble gas radionuclides are instantaneously transported into the steam flow upon entering the steam generator via the leak

- All of the noble gas radionuclides are removed via the condenser off gas system so that the entire noble gas isotope inventory enters the secondary system at the steam generator and exits at the condenser off gas
- The condenser off gas flow is accurately measured and accurately sampled
- There are no significant changes in steam or off gas flows
- Plants with mechanical vacuum pumps may have high air flow rates, which can reduce the sensitivity

5.2.3 Radionuclide Selections

The noble gases are the isotopes of choice due to their inert nature. Their use allows a simple determination of the leak rate. The isotopes of choice are as follows:

Table 5-1
Condenser Off Gas Analysis Isotopes and Half-Lives

Isotope	Half-Life
Xe-133	5.25 days
Xe-135	9.1 hours
Kr-85m	4.5 hours
Kr-88	2.84 hours
Ar-41	1.8 hours
Kr-87	1.3 hours
Xe-135m	15.29 minutes
C-11	20.39 minutes

When RCS fission gas concentrations are sufficiently high, Xe-133 (or Xe-135) provides the most reliable leakage measurement of the possible isotopes due to its long half-life and relative abundance. The other isotopes can be used, but care must be taken to minimize the effect of their shorter half-lives. Ar-41 will provide a more reliable leak rate value when RCS fission gas concentrations are low.

5.2.4 Limitations

1. The accuracy and precision of the leak rate determination using this method is dependent on the accuracy of the condenser off gas flow rate measurements.
2. The condenser off gas method measures the total primary-to-secondary leakage and does not identify the leaking steam generator.
3. In addition to the factors presented in Section 4, leak rate monitoring sensitivity will also be affected by such factors as the total condenser air inleakage (since this dilutes the concentration of the radioactive species), sample size, RCS noble gas concentrations, etc

5.2.5 Precautions

The precision and accuracy of the condenser off gas method is predominantly affected by the off gas flow rate measurement. Unless accurate mass flow measurements are made, the inaccuracies of the off gas flow rate measurement can mask most other sources of error. To increase the accuracy of this method, station personnel may consider the discussion offered above and the following additional corrections:

1. The average secondary system transport time to the condenser off gas and RCS sample line transport times to the grab samples should be evaluated. It may be necessary to correct for short-lived radionuclides, such as Ar-41 and Xe-135m. A methodology for such a correction is presented in Appendix B.
2. The parent/daughter relationship effect of iodine decay to xenon can result in errors in the leak rate determination. This effect is especially significant for the RCS sample since this will result in a non-conservative under-estimation of the leak rate. Analyzing the RCS samples as soon as possible after sampling helps to minimize the impact of parent/daughter effects. Another effect to be considered is the presence of additional xenon in the off gas from iodine decay in the steam generator bulk water. Under some circumstances, this can result in an over-estimation of the leak rate for Xe-135 and, to a lesser extent, for Xe-133. However, this error is conservative and may not be significant under most circumstances. This phenomenon is also discussed in Appendix B.
3. Steady-state conditions for the secondary noble gas concentrations and the primary noble gas concentrations are necessary to accurately determine leak rate. Changes in the RCS noble gas activity can result from power transients, changes in fuel integrity, RCS degassing operations, etc. These effects can be minimized by sampling the RCS and off gas at approximately the same time. A maximum differential of about 15 minutes is recommended during transient conditions. During steady-state conditions, the most recent sample may be used.
4. Differences in off gas and RCS sample gas volume parameters can affect the accuracy off gas method. Care can be taken to analyze the samples at approximately the same temperature and pressure to minimize the effect. If it is not possible to equalize the temperature and pressure of the samples, these parameters can be measured and corrected using the gas laws.
5. A substantial difference in the temperature and pressure conditions of the off gas activity sample and the measurement of the off gas flow rate can also introduce error. Care can be taken to ensure that the off gas activity sample and the off gas flow rate measurement are taken at approximately the same conditions.
6. Air inleakage (off gas flow) can affect the sensitivity of the method if the rate of inleakage is large. This effect can be minimized by a good air inleakage program. If a rotameter is used to measure the flow rate, it is recommended to take an average of the high-low fluctuations in readings. Some utilities sparge the condenser with nitrogen to aid in dissolved oxygen removal. If a condition exists where greater sensitivity is needed, the nitrogen sparge can be temporarily suspended during the off gas sampling to avoid this dilution. However, it should be noted that temporarily suspending sparging may increase the delay time.
7. Accurate, calibrated mass flow measuring equipment will minimize errors in leak rate calculations.

5.2.6 Example Calculation

See Appendix C for an example calculation of the primary-to-secondary leak rate using condenser air ejector monitor readings.

5.3 Leak Rate Calculations via Blowdown Analysis

5.3.1 Radionuclides and Steam Generator Blowdown

Radionuclides from the reactor coolant system enter the steam generator bulk water when a primary-to-secondary leak exists. Due to their low solubility, radiogases are quickly transported out of the steam generator bulk water into the steam. Dissolved solids and very low concentrations of radiogases remain in the steam generator bulk water. These radionuclides can be quantified in the steam generator blowdown and used to estimate primary-to-secondary leak rate and determine which steam generator is leaking. However, due to hideout, steam-water partitioning, blowdown rate, secondary system lineups, sampling uncertainties, etc., this method is the most uncertain and should be used with caution. It is suitable for gross leak rates estimates, trending, and/or identifying the leaking steam generator. See Figure 5-1 for a comparison of the accuracy of the steam generator blowdown versus the off gas method for a primary-to-secondary leak Content deleted - EPRI Proprietary Also, note in the figure how calculated leak rate for the blowdown varied depending on isotope half-life.

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Figure 5-1
Variation of Leak Rate Calculations with Radionuclide at San Onofre, 1988

5.3.2 The Basic Relationship

A relationship describing a primary-to-secondary leak in a single steam generator can be developed by developing an "activity" balance around a leaking steam generator. This relationship can be described as follows: the change in the activity concentration in a steam generator is the difference between the activity being added (leak rate and feedwater) and what is being removed (blowdown, decay and main steam). Hideout is intentionally ignored for this application. It should be noted that this assumption may result in a significant underestimation of the actual leak rate (see Section 5.3.7). Presented below is the differential equation describing this "activity" balance:

$$V_{SG}\frac{dA_{SG}}{dt}=\left[\left(F_{FW}A_{FW}\right)+\left(LRA_{RCS}\right)\right]-\left[\left(F_{S}A_{S}\right)+\left(F_{BD}A_{BD}\right)+\left(V_{SG}\lambda A_{SG}\right)\right] \qquad \textbf{Equation 5-2}$$

Where:

A_{SG}	= Activity concentration in the steam generator bulk water and blowdown (μCi/g)
A_{FW}	= Activity concentration in the feedwater (μCi/g)
A_{BD}	= Activity concentration in the steam generator blowdown (μCi/g)
A_{RCS}	= Activity concentration in the reactor coolant (μCi/g)
A_{s}	= Activity concentration in the main steam (μCi/g)
V_{SG}	= Volume of water in the steam generator (gal corrected to 77°F)
F_{FW}	= Feedwater flow to the individual steam generator (gpd corrected to 77°F)
LR	= Primary-to-secondary leakage (gpd)
F_{s}	= Main steam flow from the individual steam generator (gpd corrected to 77°F)
F_{BD}	= Blowdown flow rate (gpd corrected to 77°F)
λ	= Decay constant (reciprocal days)
t	= Time (days)

5.3.3 The Simplified Transient

The solution to the transient relationship is complicated and relies on obtaining precise results of a number of activity samples and flow rates at two discrete times. Due to the inherent errors and assumptions in the data used for the analysis, the benefits achieved in using this solution will be offset by a significant error associated with the solution. However, a number of assumptions can be made to simplify the transient solution. If it is assumed that all activity in the steam returns in the feedwater ($F_SA_S = F_{FW}A_{FW}$) and that at time equals zero (t_0) there is no primary-to-secondary leak rate or the activity in the steam and the feedwater are zero (A_S=0 and A_{FW}=0) and that at time equals zero (t_0) there is no primary-to-secondary leakage, the simplified transient relationship becomes:

$$LR = \frac{A_{SG}\left(F_{BD} + \lambda V_{SG}\right)}{A_{RCS}\left(1 - e^{-(\lambda+\beta)t}\right)}$$ **Equation 5-3**

Where all variables are as defined in Equation 5-2 and:

β = F_{BD}/V_{SG}, the purification rate constant (reciprocal days, assumes blowdown is discharged or that blowdown demineralizers or condensate polishers have a high removal efficiency)

The assumption above have limited application with condensate polishing is in-service. Each plant should evaluate the significance of this assumption before using the simplified transient solution presented above.

5.3.4 The Steady-State Solution

The transient relationship simplifies to the following during steady-state conditions:

$$LR = \frac{A_{SG}\left(F_{BD} + \lambda V_{SG}\right)}{A_{RCS}}$$ **Equation 5-4**

All the variables are as defined in Equation 5-2. It was assumed that all activity in the steam returns in the feedwater ($F_S A_S = F_{FW} A_{FW}$) and that at time equals zero (t_0) there is no primary-to-secondary leakage or the activity in the steam and the feedwater are zero (A_S=0 and A_{FW}=0) and that at time equals zero (t_0) there is no primary-to-secondary leakage

If this assumption is not valid, then the following more general equation can be used:

$$LR = \frac{F_{d1}\left(A_{d1b} - A_{d1a}\right) + F_{d2}\left(A_{d2b} - A_{d2a}\right) + \lambda V_{SG} A_{SG}}{A_{RCS}}$$ **Equation 5-5**

Where:

F_{d1} = Flow rate through blowdown demineralizer (gpd corrected to 77°F)

F_{d2} = Flow rate through condensate polisher (gpd corrected to 77°F)

A_{d1b} = Activity concentration before blowdown demineralizer (µCi/g)

A_{d1a} = Activity concentration after blowdown demineralizer (µCi/g)

A_{d2b} = Activity concentration before condensate polisher demineralizer (µCi/g)

A_{d2a} = Activity concentration after condensate polisher demineralizer (µCi/g)

A_{RCS} = Activity concentration in the reactor coolant (µCi/g)

This equation assumes steady-state leak rate and secondary system conditions, and that system leakage, system hideout and other removal terms are small compared to the demineralizer resin. If these assumptions are not valid, a more detailed relationship should be developed.

5.3.5 Radionuclide Selections

In general, any radionuclide that can be accurately identified in the steam generator blowdown and the RCS can be used to estimate primary-to-secondary leak rate. However, for simplicity and relative accuracy, radionuclides are chosen according to the following criteria:

1. A moderately short half-life to allow separation from old leaks and to reduce time to equilibrium
2. Be water soluble and have a very small steam-water partitioning factor (not readily transported out of the steam generator in the steam)
3. Be relatively abundant in the RCS to provide sensitivity and be easily measurable by gamma spectroscopy

The isotopes generally chosen for quantification are as follows:

Table 5-2
Blowdown Analysis Isotopes and Half-Lives

Isotope	Half Life
Na-24	15 hours
I-131	8 days
I-132	2.3 hours
I-133	21 hours
I-134	53 minutes
I-135	6.6 hours
Cs-138	32 minutes

5.3.6 Limitations

1. The steam generator blowdown method provides only an estimate of the leak rate (accuracy is limited). It is suitable for trending purposes and may be used for identification of which steam generator is leaking.
2. In addition to the factors listed in Section 5.3.7, sensitivity is affected by the steam generator blowdown rate.

5.3.7 Precautions

1. Many factors can affect the error associated with the steam generator blowdown method such as steam carry over of the radionuclide, partitioning effects, hideout/return, chemical reactions, leak location, etc. For example, iodine readily reacts with sludge metals and is subject to hideout. However, it may be possible in some circumstances to obtain leak rate which correlates with the more reliable methods, taking these factors into consideration and making the applicable corrections.
2. The steam generator blowdown sample itself may not truly reflect its own generator bulk water, due to deposition of solids within the sample line, downcomer/feedwater mixing, etc. Each plant should evaluate the specific design for transport time and determination of a representative sample of the steam generator bulk water.,
3. A steady-state condition of the secondary system is essential to avoid gross errors in the estimate of leak rate. A change in steam generator blowdown rate, for example, can introduce error into the result until a new steady-state is achieved.
4. Inaccuracies in the measurement of the necessary flow rates can introduce errors.
5. Inaccuracies in the steam generator liquid volume also introduce error. This error is minimized by maximizing steam generator blowdown rate.

5.4 Leak Rate Calculations via Tritium

5.4.1 Introduction

Tritium from the reactor coolant enters the SGs and secondary system during normal operation via diffusion through SG tubing and when a primary-to-secondary leak exists. All of the tritium can be assumed to be in the form of tritiated water. The advantages in quantifying leak rate via tritium over other methods are listed below:

- Do not have to consider ion exchange effects when considering steady-state.
- Do not have to consider hideout and hideout-return effects.
- Do not have to account for liquid/steam partitioning.
- Do not have to account for concentration effects in the blowdown for SGs.
- Sampling errors are minimized.
- Is universally applicable to OTSGs and RSGs.
- The radiochemistry analysis is precise, accurate, and specific for tritium.
- The half-life of tritium is long (12.3 years), therefore decay considerations are unnecessary. This makes it especially advantageous for tracking small leaks.
- It should be noted that the typical tritium concentration in the primary side of a CANDU pressurized heavy water reactor is in the order of 1000 μCi/mL thus making tritium the preferred method of leak detection.

Because of these factors, the tritium method is recommended to be used to validate the condenser off gas and SG blowdown methods when the circumstances permit. This assumes that the RCS tritium is at the necessary concentration, the situation is not urgent as during a rapidly propagating leak, and the tritium is not at steady-state in the secondary which requires accurate makeup rates.

The major disadvantage of the tritium method is an effect of the long half-life of tritium. This makes system leakage the only practical removal mechanism which requires a lengthy time period, relative to most other isotopes used for leak rate determinations, to reach equilibrium.

This reduces the sensitivity of the method with respect to identifying a new leak in the initial stages. *It also affects the leak rate calculation due to the lag time in the secondary tritium concentrations following changes in the RCS tritium concentration.*

The following sections deal with calculating the leakage prior to steady-state (see Sections 5.4.2.1 and 5.4.2.2) and after steady-state is achieved (see Section 5.4.2.3) in the secondary system. The leak rate prior to steady-state can be determined, under some circumstances, in a reliable manner with knowledge only of the secondary system volume. If the required conditions do not exist, it is necessary to accurately know the makeup rate. Because of the half-life of tritium, the time to steady-state may be 2-3 weeks with a maximum makeup rate of 50 gpm. Plant designs with higher makeup rates (blowdown is directed to waste) of approximately 100 gpm or greater may achieve steady-state in as few as 3 days. After steady-state, the tritium concentration in the secondary system becomes a function of the makeup rate, which must be accurately known in order to reliably calculate leak rate.

5.4.2 The Tritium Calculations

5.4.2.1 Tritium Calculation Prior to Steady-State

$$LR = \frac{\left(A_{s2} - A_{s1} e^{-\left(\frac{Mu}{V_S}+\lambda\right)\Delta t}\right)\left(Mu + \lambda V_s\right)}{A_{RCS}\left(1 - e^{-\left(\frac{Mu}{V_S}+\lambda\right)\Delta t}\right)} - \frac{A_{Mu} Mu}{A_{RCS}}$$

Equation 5-6

Where:

LR	= Leak Rate, gpd
A_{s2}	= Activity concentration of tritium in the secondary coolant at t_2 (µCi/g)
A_{s1}	= Activity concentration of tritium in the secondary coolant at t_1 (µCi/g)
V_S	= Secondary system mass expressed as equivalent room-temperature volume (gal)
Δt	= $(t_2 - t_1)$ = the difference of the sample times (days)

A_{RCS}	= Activity concentration of tritium in the primary coolant (μCi/g)
Mu	= Makeup rate to secondary system (gpd)
λ	= Decay constant for tritium (1.54E-4 reciprocal days)
A_{Mu}	= Activity concentration of tritium in the makeup water (μCi/g)

When the following assumptions are valid:

- The RCS tritium is at a concentration high enough to permit secondary leak rate measurement and is constant
- The mass of the secondary system is accurately known
- The secondary makeup rate is relatively constant

The above equation does not consider tritium diffusion through the steam generator tubing. References [1] and [2] note that uncertainties in the calculation of the rate of hydrogen diffusion, and by analogy, tritium diffusion, could lead to variations in the calculated diffusion rate by up to approximately an order of magnitude. Under operating conditions that are typical at some units, it is plausible that this relatively large degree of uncertainty could lead to the calculation of a tritium diffusion rate that is roughly comparable to the tritium contribution of a 5 gpd leak. In such an instance, the 5 gpd leakage threshold would not be identified until the leak rate had increased above 5 gpd. It is likely that this issue would be best addressed on a plant-specific basis. Such an effort would require an operational determination of the tritium diffusion rate in the confirmed absence of leakage as it is unlikely that the steady-state tritium diffusion rate at a given unit would vary significantly over time.

5.4.2.2 Tritium Calculation Simplified for Specific Conditions (Prior to Steady-State)

If the activity concentration of tritium in the makeup water is much less than in the secondary coolant, i.e., $A_{Mu} << A_{s1}$ and A_{s2}, and if Mu is >> λVs, then Equation [5-6] reduces to:

$$LR = \frac{\left(A_{s2} - A_{s1} e^{-\left(\frac{Mu}{V_S}+\lambda\right)\Delta t} \right) Mu}{A_{RCS}\left(1 - e^{-\left(\frac{Mu}{V_S}+\lambda\right)\Delta t} \right)}$$

Equation 5-7

If the leak is of the magnitude that the increase from A_{S1} to A_{S2} is large during a relatively short time period, (approximately 24 hours) and the secondary makeup rate is small during the period compared to the system volumes, then:

$$LR = \frac{V_s\left(A_{s2} - A_{s1}\right)}{A_{RCS}\Delta t}$$

Equation 5-8

5.4.2.3 Tritium Calculation after Steady-State

When steady-state is achieved, the following equation can be used to calculate leakage:

$$LR = \frac{A_s Mu}{A_{RCS}}$$ **Equation 5-9**

Where:

LR = Leak Rate (gpd)

A_s = Tritium activity concentration in the secondary coolant (μCi/g)

A_{RCS} = Tritium activity concentration in the primary coolant (μCi/g)

Mu = Makeup rate to secondary system (gpd corrected to 77°F)

When the following assumptions are valid:

- The RCS tritium is at the necessary concentration.
- The only source of tritium is from the primary-to-secondary leakage.
- The secondary makeup rate is relatively constant.

5.4.3 Limitations

1. Tritium provides total system leak rate and cannot be used to determine which steam generator is leaking. However, after the leaking steam generator is isolated, tritium samples from the steam generator can be used to provide an accurate measure of leak rate from the leaking steam generator using Equation [5-7] and substituting the steam generator volume for the secondary system volume.
2. A survey of plants indicated very similar tritium concentrations in secondary systems. Typical values were in the range of 5E-6 to 5E-5 μCi/g and varied with primary system tritium concentration. It would seem unlikely that random manufacturing imperfections could result in such similar concentrations. A sample calculation performed for Revision 4 demonstrated that tritium diffusion can produce tritium concentrations in the secondary system in the range of the actual measured concentrations [2]. The work documented in Reference [2] shows that when significant primary-to-secondary leakage is present, the contribution of tritium diffusion is minor and can be ignored. In the absence of primary-to-secondary leakage, it is likely that some, and possibly a major portion, of the tritium in nuclear plant secondary systems is due to diffusion of tritium through nickel alloy SG tubing.

3. Sensitivity of the method depends on the following interrelated factors:
   - Tritium concentration in the RCS: This is not directly related to fuel condition like the other isotopes, but does depend on growth of tritium concentration in the RCS due to other factors. This is usually only an issue during periods of significant dilution evolutions of the RCS and for plants in early core life when there is little tritium in the RCS. The tritium method will provide the best sensitivity when the RCS tritium is 0.01-0.1 μCi/g or greater.
   - Time elapsed from initiation of the leak for buildup of tritium concentration in the secondary system towards the equilibrium value
   - Secondary system leak rate (system makeup rate): Note that this can be very high and erratic for PWRs that use condensate to rinse condensate polishers.
   - Introduction of errors due to multiple-unit water transfers where water is freely transferred between units
   - Diffusion of tritium through steam generator tubing leading to a low level of tritium in the secondary system
   - Figure 5-2 gives guidance for approximate time between samples for detection of leak rate at different primary system tritium levels. The figure is provided as a starting point and should be modified in accordance with plant experience.

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Figure 5-2
Leak Rate Sensitivity for Tritium

5.4.4 Precautions

1. Care must be taken in the applicable calculations to ensure that the makeup rate is reflective of the entire time period between samples or corrected for the additional dilution. Although the makeup rate at two time points may be the same, it may not necessarily have been constant over the entire time period.
2. The entire applicable secondary volume must be accounted for in the calculation. This must include the mass in the steam generators, piping, components, and recirculation masses such as condensate storage tanks.
3. Care should be exercised in the radiochemistry counting of the tritium samples due to the possible interferences with counting other low-energy betas. If possible, samples should be drawn from a demineralizer effluent.
4. Non-steady-state errors can occur after trips or down-powers where the RCS tritium is rapidly and measurably reduced. The resulting leak rate will erroneously indicate a large increase during the time period required for the secondary tritium to return to steady-state, unless the pre-dilution RCS tritium value can be reliably used. This time period can be several weeks in length and the error can be large; therefore, another method should be evaluated as the primary method or as a corroboration method during this period.

5.5 Leak Rate Evaluation via Other Methods

5.5.1 Introduction

There are other qualitative and quantitative methods to detect and determine primary-to-secondary leakage. Qualitative methods can be used for a rapid determination of the presence of a leak and identification of the leaking steam generator.

5.5.2 Leak Rate Quantification with Leaks from Multiple Steam Generators

In many plant designs, the ability to determine actual per steam generator leak rate is complicated by the cross contamination of activity from one steam generator to another. Plants with N-16 monitors or other main steam line monitors with sufficient sensitivity may be able to estimate individual steam generator leak rate by comparing the individual steam line monitor readings. Plants without N-16 monitors may wish to quantify the leak rate for individual steam generators. Activity concentration in the steam can also be used to estimate the relative leak rate from each generator. A qualitative determination can be accomplished by the use of a portable multiple-channel analyzer (MCA) to measure N-16 at the steam line of each steam generator and comparing the results to the total leak rate determined using methods such as condenser off gas monitors or sampling.

5.5.2.1 Recirculating Steam Generators

To calculate leak rate for individual steam generators, the activity in each generator blowdown can be determined. With one or more leaking steam generators, the activity will be distributed between all steam generators. The leaking steam generator will have a higher activity. With more than one steam generator leaking, a ratio can be used to determine a crude estimate of the relative distribution.

$$LR_{SG,i} = \frac{A_{SG,i} LR_T}{A_{SG,T}}$$ **Equation 5-10**

Where:

$LR_{SG,i}$ = Leak rate through steam generator, *i* (gpd)

LR_T = Total combined leak rate through all steam generators (gpd)

$A_{SG,i}$ = Activity concentration in steam generator, *i* (μCi/g)

$A_{SG,T}$ = Sum of all activity concentrations in all steam generators (μCi/g)

The individual steam generator activity used can be from a total gross beta analysis or individual isotopes, such as I-131, I-133, I-135 or Na-24. The accuracy of this estimate is dependent on the amount of moisture carryover, blowdown routing, and other site-specific factors. Since the activity from a single leaking steam generator will be distributed, to some extent, between all steam generators depending on site-specific factors, this technique may underestimate leak rate in the leaking steam generator and indicate leak rate in some non-leaking steam generators.

5.5.3 Main Steam Sample Method for Leak Rate Quantification

5.5.3.1 Discussion

The main steam sample method is very similar to the condenser off gas method described in Section 5.2 because it involves collection and analysis of radioactive noble gases. Some plants have the capability of sampling the main steam. These plants (both OTSGs and RSGs) can use the main steam total gas sample and the noble gas isotope selection from Section 5.2 to estimate leak rate. The leak rate is calculated as follows:

$$LR = \frac{2.88 A_{STM} F_{STM}}{A_{RCS}}$$ **Equation 5-11**

Where:

LR = Leak rate (gpd)

A_{STM} = Activity concentration of the isotope in the main steam sample (μCi/g)

F_{STM} = Flow rate of the main steam at the specific power level when the sample was taken (lbm/hr)

A_{RCS} = Activity concentration of the radionuclide in the RCS sample (μCi/g)

2.88 = gpd per lbm/hr, conversion constant which includes:

1 day per 24 hours
3785 ml per gallon
1 g per ml
1 lbm per 453.6 grams

5.5.3.2 Limitations

1. The accuracy and precision of the main steam total gas sample method is affected by the difficulties of obtaining a representative steam sample. Also, the main steam is not as concentrated as the condenser off gas and this method is therefore not as sensitive. The results may be more qualitative than quantitative. This method could be correlated with more accurate methods such as tritium.
2. The collected steam sample should be carefully drawn and sample container secured to avoid loss of the gases to be counted.
3. Equation [5-11] assumes no activity from the chosen isotope in the steam generator feedwater. If this assumption is not valid, the equation should be corrected for the feedwater contribution to main steam.
4. Precautions 1, 2 and 3 in Section 5.2.5 from the condenser off gas method may also apply and should be evaluated.

5.5.4 Steam Generator Blowdown Cation Columns (or Resin-Impregnated Filters)

5.5.4.1 Sampling Method

Steam generator blowdown cation columns or samples collected on resin-impregnated filter papers in corrosion product samplers provide a convenient measurement of primary-to-secondary leak rate. They can be used as a rapid determination of which steam generator is leaking by using a portable radiation detection instrument to determine which steam generator cation column contains the most activity. If the area background activity will not allow this determination, then the columns may be removed to a low background area and monitored there, if this can be accomplished quickly enough to be of value.

5.5.4.2 Calculation

A qualitative estimate of the leak rate may be determined using cation columns. This is especially suitable for tracking small leaks on the order of approximately 1-2 gpd or less. This is accomplished by measuring the flow through the cation column for the time necessary to obtain suitable isotope concentrations for γ spectroscopy. The resin must be quantitatively transferred to a suitable container for identification of cation radioisotopes by γ spectroscopy. The leak rate can then be estimated by the following formula:

$$LR = \frac{1440 \frac{A_R}{V} F_{BD}}{A_{RCS}}$$

Equation 5-12

Where:

LR	= Leak rate (gpd)
A_R	= Total curie content of the isotope in the resin (μCi)
A_{RCS}	= RCS activity of isotope of interest (μCi/g)
V	= Volume of sample passed through the cation column (ml)
F_{BD}	= Steam generator blowdown flow (gpm corrected to 77°F)
1440	= min/day per g/ml , conversion constant which includes the following: 60 minutes per hour 24 hours per day

This calculation is subject to the same assumptions, limitations, and precautions as in Section 5.3. In addition, it assumes that the radioisotope selected has a long decay half-life compared to the steam generator blowdown half-life and sampling period. However, it should not be necessary to make the corrections suggested for other isotope conditions in the referenced appendices. Additional error or variance can be introduced by the uncertainties of measuring resin volumes and the method chosen for determining the volume of sample. For example, if the median flow rate is used, this will obviously introduce error. If the sample period is lengthy, variations in flows, such as blowdown, will also introduce errors. But, it is judged not to be beneficial to correct for these errors. This method is used to track very small leaks where even large variances or errors are not a significant concern, especially since it can be correlated with the tritium analysis.

5.5.4.3 Selection of Radionuclides

Because the time periods to obtain suitable concentrations on the resin for γ spectroscopy can be lengthy (up to 30 days), Cs-137 or Cs-134 are the preferred radionuclides. The 30 year half-life of Cs-137 or the 2.06 year half-life of Cs-134 makes decay correction unnecessary and eliminates this error. If the time period is short enough, other radionuclides may be utilized (such as Na-24), but it may be necessary to correct for decay.

5.5.4.4 Alternate Method

Another estimate can be obtained by passing a known volume of steam generator sample (20-40 liters) through a column with a suitable volume (~50 ml) of mixed bed HOH (hydrogen ion—hydroxyl ion) form resin. After quantitatively transferring the resin for γ spectroscopy, the activity for a specific isotope can be calculated as follows:

$$A_{SG} = \frac{A_R}{V} \qquad \textbf{Equation 5-13}$$

Where:

A_{SG} = Activity concentration of the steam generator sample (μCi/ml)

A_R = Total curie content of the resin (μCi)

V = Volume of the sample passed through the resin (ml)

The leak rate can then be calculated using the calculations and radionuclide selections in Section 5.3 utilizing the same assumptions, limitations, and precautions.

5.5.5 Steam Generator Blowdown Cleanup Systems

Due to the requirement for activity monitoring of any material "free released" from the controlled area, plants with steam generator blowdown resin cleanup system and/or filters can obtain an early indication of a small primary-to-secondary leak if these materials become contaminated. However, this is a qualitative indication only and a quantitative leak rate determination should be accomplished with other methods.

5.5.6 Condensate Polisher Resin Analyses

Some slightly volatile species that enter the secondary system via primary-to-secondary leak rate (e.g., radioiodines) will exit the steam generator via the steam and partially condense with condensate. These isotopes will be collected on the ion exchange resins in the condensate polishers. Sampling and analyses of these resins can be used to detect/confirm (but not quantify) the existence of suspected primary-to-secondary leak rate. This method will not be further discussed due to its uncertainties (from non-representative sampling, unknown decay times, etc.).

5.6 Leak Rate Calculations in Non-Operating Modes

Appendix D discusses the need to monitor leak rate in non-operating modes and provides a discussion of the methods used for such monitoring. As indicated in Appendix D, the normal method used is tritium. The basic equation used for monitoring primary-to-secondary leak rate in non-operating modes is the same as Equation 5-8:

$$LR = \frac{V_S(A_{s2} - A_{s1})}{A_{RCS}(\Delta t)}$$

Equation 5-14

Where:

LR = Leak rate (gpd)

V_s = Active volume of secondary (gal corrected to 77°F)

A_{s2} = Activity concentration in secondary at time, t_2 (μCi/g)

A_{s1} = Activity concentration in secondary at time, t_1 (μCi/g)

A_{RCS} = Activity concentration in primary (μCi/g)

Δt = $(t_2 - t_1)$ = the difference of the sample times (days)

Note: The secondary volume should be adjusted to reflect the portion of the system being sampled and in contact with the potentially leaking generator(s) to account for the situations when the full secondary volume is not involved.

This equation results in some error unless the steam generator volume is constant and there is no blowdown. A more complex methodology, which can accommodate changes in level or blowdown, is presented in Appendix D.

5.7 References

1. *Hydrogen Loss from the Reactor Coolant System during a Loss-of-Letdown Event,* Dominion Engineering, Inc., Reston, VA: 2007. C-5547-00-01.
2. *Tritium Diffusion Assessments*, Dominion Engineering, Inc., Reston, VA: 2009. M-5593-00-04, Revision 3.

6
REQUIREMENTS AND RECOMMENDATIONS

In accordance with the SGMP Administrative Procedure, Revision 2, this section clearly identified mandatory and shall (or needed) requirements. Recommendations (or best practice) elements of this guideline document are included as well.

Section 6 summarizes the requirements. Refer to the appropriate section in Section 3 for detailed explanation of the requirements.

Note that "*" means "If unable to determine leakage from individual steam generators, the total leakage is assumed to be coming from one steam generator".

6.1 Programmatic Requirements

Category	Requirement	SGMP Designation	Section
Programmatic	Content deleted - EPRI Proprietary	**Mandatory**	3.3.1
Programmatic	Content deleted - EPRI Proprietary	**Mandatory**	3.3.1

6.2 Primary-to-Secondary Leak Rate Monitoring Program

Category	Requirement	SGMP Designation	Section
Mode 3 and 4 Monitoring (Heatup or Cooldown)	Content deleted - EPRI Proprietary	**Shall**	3.4.1
Mode 3 and 4 Monitoring (Heatup or Cooldown)	Content deleted - EPRI Proprietary	Recommended	3.4.1
Mode 1 and 2 Monitoring Non-Steady-State	Content deleted - EPRI Proprietary	**Shall**	3.4.2.1
Steady-State Power Operation	Content deleted - EPRI Proprietary	**Shall**	3.4.2.2
Steady-State Power Operation	Content deleted - EPRI Proprietary	**Shall**	3.4.2.2

Category	Requirement	SGMP Designation	Section
Steady-State Power Operation	Content deleted - EPRI Proprietary	**Mandatory**	3.4.2.2

6.3 Primary-to-Secondary Leak Rate Actions (Modes 3 and 4)

Category	Requirement	SGMP Designation	Section
Modes 3 and 4	Content deleted - EPRI Proprietary	**Shall**	3.5.1
Modes 3 and 4	Content deleted - EPRI Proprietary	Recommended	3.5.1

6.4 Primary-to-Secondary Leak Rate Actions – Mode 1 and 2 Non-Steady-State and Steady-State Power Operations – No Available Continuous Radiation Monitor

Category	Requirement	SGMP Designation	Section
No Available Continuous Radiation Monitor ENTRY	Content deleted - EPRI Proprietary	**<u>Mandatory Entry</u>**	3.5.2.1
No Available Continuous Radiation Monitor	Content deleted - EPRI Proprietary	**<u>Mandatory</u>**	3.5.2.1
No Available Continuous Radiation Monitor	Content deleted - EPRI Proprietary	Recommended	3.5.2.1

6.5 Primary-to-Secondary Leak Rate Actions – Mode 1 and 2 Non-Steady-State and Steady-State Power Operations – Continuous Radiation Monitor – Rate of Change Methodology Actions

Category	Requirement	SGMP Designation	Section
Action Level 3 ENTRY	Content deleted - EPRI Proprietary	**<u>Mandatory Entry</u>**	3.5.2.2.1
Action Level 3	Content deleted - EPRI Proprietary	**<u>Mandatory</u>**	3.5.2.2.1
Action Level 3	Content deleted - EPRI Proprietary	**Shall**	3.5.2.2.1
Action Level 3	Content deleted - EPRI Proprietary	Recommended	3.5.2.2.1

Category	Requirement	SGMP Designation	Section
Action Level 2 ENTRY	Content deleted - EPRI Proprietary	**Mandatory Entry**	3.5.2.2.2
Action Level 2	Content deleted - EPRI Proprietary	**Mandatory**	3.5.2.2.2
Action Level 2	Content deleted - EPRI Proprietary	**Shall**	3.5.2.2.2
Action Level 2	Content deleted - EPRI Proprietary	Recommended	3.5.2.2.2

Category	Requirement	SGMP Designation	Section
Action Level 1 ENTRY	Content deleted - EPRI Proprietary	**Shall Entry**	3.5.2.2.3
Action Level 1	Content deleted - EPRI Proprietary	**Shall**	3.5.2.2.3

Category	Requirement	SGMP Designation	Section
Action Level 1	Content deleted - EPRI Proprietary	Recommended	3.5.2.2.3
Increased Monitoring ENTRY	Content deleted - EPRI Proprietary	**Shall Entry**	3.5.2.2.4
Increased Monitoring	Content deleted - EPRI Proprietary	**Shall**	3.5.2.2.4

Category	Requirement	SGMP Designation	Section
	Content deleted - EPRI Proprietary		
Increased Monitoring		Recommended	3.5.2.2.4

6.6 Primary-to-Secondary Leak Rate Actions – Mode 1 and 2 Non-Steady-State and Steady-State Power Operations – Continuous Radiation Monitor – Constant Leak Rate Methodology

Category	Requirement	SGMP Designation	Section
Action Level 3 ENTRY	Content deleted - EPRI Proprietary	**Mandatory Entry**	3.5.2.3.1
Action Level 3	Content deleted - EPRI Proprietary	**Mandatory**	3.5.2.3.1
Action Level 3	Content deleted - EPRI Proprietary	**Shall**	3.5.2.3.1
Action Level 3	Content deleted - EPRI Proprietary	Recommended	3.5.2.3.1
Action Level 2 ENTRY	Content deleted - EPRI Proprietary	**Mandatory Entry**	3.5.2.3.2
Action Level 2	Content deleted - EPRI Proprietary	**Mandatory**	3.5.2.3.2

Category	Requirement	SGMP Designation	Section
Action Level 2	Content deleted - EPRI Proprietary	**Shall**	3.5.2.3.2
Action Level 2	Content deleted - EPRI Proprietary	Recommended	3.5.2.3.2

Category	Requirement	SGMP Designation	Section
Action Level 1 ENTRY	Content deleted - EPRI Proprietary	**Shall Entry**	3.5.2.3.3
Action Level 1	Content deleted - EPRI Proprietary	**Shall**	3.5.2.3.3

Category	Requirement	SGMP Designation	Section
Action Level 1	Content deleted - EPRI Proprietary	Recommended	3.5.2.3.3
Increased Monitoring ENTRY	Content deleted - EPRI Proprietary	**Shall Entry**	3.5.2.3.4
Increased Monitoring	Content deleted - EPRI Proprietary	**Shall**	3.5.2.3.4

Category	Requirement	SGMP Designation	Section
	Content deleted - EPRI Proprietary		
Increased Monitoring		Recommended	3.5.2.3.4

6.7 Primary-to-Secondary Leak Rate Actions – Power Transients with Known Primary-to-Secondary Leakage

Category	Requirement	SGMP Designation	Section
Power Transients with Known Primary-to-Secondary Leakage	Content deleted - EPRI Proprietary	Recommended	3.5.3

A
EFFECTS OF PRIMARY SIDE pH ON LEAKAGE FROM SCC CRACKS

The EPRI Primary Water Chemistry Guidelines [1] provide guidance regarding primary water pH control for maintenance of pressure boundary integrity, fuel integrity, and system performance. One collateral consequence of chemistry changes is that changes in the primary coolant pH may affect primary-to-secondary leakage without affecting flaw size, thus altering the relationships between leakage and burst pressure discussed in Section 2.2.1. This section reviews the experience relating primary pH to leak rate at the following units/utilities:

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Additionally, three of the four operating experience cases discussed in Section 2.2.1.5 Content deleted - EPRI Proprietary were evaluated to determine the effect of primary side pH on the extent to which the operating experience supports the models developed in Section 2.2.1.

Section 2.3.4 presents the conclusions reached concerning the effect of primary coolant pH on primary-to-secondary leak rate and leak rate limits.

A.1 Ringhals Data

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Figure A-1
Primary Side pH and Leakage as a Function of Time, Ringhals Unit 4

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Figure A-2
Leakage as a Function of pH, Ringhals Unit 4

A.2 EDF Data

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A.3 Tihange Data

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Figure A-3
Primary Side pH and Leak rate as a Function of Time, Tihange Unit 2

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Figure A-4
Leak Rate as a Function of pH, Tihange Unit 2

A.4 Operating Experience Example Cases

Primary side pH data were obtained for three of the described in Section 2.2.1.5 [6–8]. These data indicated the following pH values:

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A.5 Conclusions

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A.6 References

1. *Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1, Revision 6*, EPRI, Palo Alto, CA: 2007. 1014986.
2. B. Bengtsson and S. Larsson, "Ringhals NPP Evaluation and Experience of pH-Control Regarding Radiation Levels and SG Integrity," Attachment to the Meeting Report of the EPRI Primary Water Chemistry Guidelines Committee July 2006 Meeting, 2006.
3. Caramel and Menet, Study of the Behavior of Primary/Secondary Leaks Changing at the End of the Cycle: Influence of Physico-Chemical Parameters, EDF, Saint Denis, France, 1985 EDF D 5001/CRL/R 85 562.
4. FP. Calle, M. Bonnet, M. Plagnol, Primary-Secondary Leak Rates through Tensile Stress Corrosion Cracks Developed in PWR Steam Generator Tubes, EDF, Saint Denis, France, 1986. EDF D 5002/CLE/R 86 427.
5. E. Girasa (Laborelec), e-mail to C. Marks (DEI), March 12, 2009 (with attachments)
6. D. Rickertsen (SNC), e-mail to C. Marks (DEI), March 3, 2009.
7. K. Johnson (Duke), e-mail to C. Marks (DEI), March 3, 2009.
8. J. Stevens (Luminant), e-mail to C. Marks (DEI), March 3, 2009.

B
CONDENSER OFF GAS CORRECTIONS

B.1 Condenser Off Gas (COG) Transport Decay Effects

The average transport time of radiogas from the steam generator to the condenser off gas sample collection point may be significant relative to the half-lives of certain radiogases. This transport time is plant-specific (generally longer at units with feedwater deaerators) and has been known to vary from 3 to 30 minutes. When using a radiogas of a relatively short half-life (e.g., Ar-41) the decay associated with the transport time may be significant. The plant-specific radiogas average transport time may be estimated as follows:

$$T = \frac{\ln\left(\frac{A_{g,Xe-133}}{A_{g,Ar-41}}\right) - \ln\left(\frac{A_{RCS,Xe-133}}{A_{RCS,Ar-41}}\right)}{\lambda_{Ar-41} - \lambda_{Xe-133}}$$

Equation B-1

Where:

T = Transport time (minutes)

$A_{g,x}$ = Activity concentration of identified radionuclide in off gas (μCi/cc)

$A_{RCS,x}$ = Activity concentration of identified radionuclide in RCS (μCi/g)

λ_x = Decay constant of identified radionuclide ($0.693/t_{1/2}$) (minutes^{-1})

It may be appropriate to make a decay correction if a 20 minute transit time exists and a radionuclide such as Ar-41 (half-life of 110 minutes) is being used to quantify primary-to-secondary leak rate. Each plant should evaluate the significance of this transport time relative to the radionuclide used in the primary-to-secondary leak rate calculation. If necessary, the activity concentration of the identified radionuclide in the off gas may be corrected for transport time using the methodology provided in the following discussion.

Substitute $A_{g(c)}$ for A_g in the leak rate calculation in Equation B-1 according to:

$$A_{g(c)} = \frac{A_g}{e^{-\lambda T}}$$

Equation B-2

Where:

$A_{g(c)}$	= Activity of the noble gas radionuclide corrected for decay time (µCi/cc)
A_g	= Activity of the noble gas radionuclide as measured in the sample (µCi/cc)
λ	= 0.693 divided by the half-life of the noble gas radionuclide (reciprocal time)
T	= Transport time (in the same time units as the half-life)

It should be noted that under some circumstances it may be necessary to correct primary coolant samples for delay times.

B.2 Parent/Daughter Relationship Effects

Because radio-isotopes decay to other radio-isotopes, the activity of a particular isotope may be enhanced by generation or depleted by decay. The decay reactions coupling pairs of radio-isotopes are referred to as parent/daughter relationships. A number of radiogas radionuclides which are typically used for primary-to-secondary leak rate quantification have parent/daughter relationships, which may affect the calculated leak rate. For example, I-135 present in the steam generator bulk water decays to Xe-135 which is then removed from the steam generators with the main steam. This can result in approximately a 10-20% error in the conservative direction. The actual magnitude of the error will depend on the I-135/Xe-135 ratio in the reactor coolant and the blowdown rate and disposition. A similar relationship exists for I-133 and Xe-133, but the associated error is much smaller. Since these errors are in the conservative direction, it is not usually considered necessary to correct the result.

If deemed necessary, this error can be evaluated by comparing the leak rate value with other radionuclides, such as Ar-41 and Kr-85m, for which the parent/daughter effect does not occur.

A parent/daughter effect is also observed when measuring dissolved xenon in a reactor coolant sample due to the decay of iodine in the reactor coolant to xenon. If a reactor coolant sample is analyzed promptly, this effect is minimized and need not be considered. However, since this error is in the non-conservative direction (i.e., it can result in underestimation of primary-to-secondary leak rate), care should be taken to minimize this error.

As part of its overall program development, each plant should evaluate and understand the significance of these parent/daughter relationships and incorporate them into the condenser off gas primary-to-secondary leak rate calculation if it is considered significant. Section B.2.1 presents a methodology example that can be used to account for the decay of iodines to xenons.

B.2.1 Decay Correcting for Iodine Activities

Primary-to-secondary leak rate calculations based on COG xenon gas isotopic activity concentrations may be corrected for decay contributions from steady state levels of I-133 and I-135 in the secondary system as follows:

$$LR \approx \frac{10773 A^{M}_{COG,j} F_{COG}}{\left[A_{RCS,j} + \left(\frac{A_{RCS,i} f_j \lambda_j}{\lambda_i + \beta_{DIS}} \right) \right]}$$

Equation B-3

Where:

LR	= *Leak Rate (gpd)*
$A^{M}_{COG,j}$	= Activity of COG xenon noble gas isotope, j (µCi/cc)
$A_{RCS,j}$	= Activity of RCS xenon isotope, j (µCi/g)
$A_{RCS,i}$	= Activity of RCS parent iodine isotope, i (µCi/g)
F_{COG}	= Total COG flow rate for evaluation period (cfm)
f_j	= Beta decay branching fraction of parent iodine isotope into meta-stable or ground state daughter xenon isotope, j, [~0.0288 (Xe-133m), 0.9712 (Xe-133), 0.165 (Xe-135m) and 0.835 (Xe-135)]
λ_j	= Decay constant of daughter xenon isotope, j [3.663E-6 (Xe-133m), 1.530E-6 (Xe-133), 7.551E-4 (Xe-135m) and 2.116E-5 (Xe-135) (sec^{-1})]
λ_i	= Decay constant of parent iodine isotope, i [9.257E-6 (I-133) and 2.931E-5 (I-135) (sec^{-1})]
β_{DIS}	= Steam generator blowdown discharge constant ≈ blowdown mass-volume discharge rate ÷ SG mass-volume, (sec^{-1})
F_{COG}	= Total COG flow rate for evaluation period (cfm)
10773	= Unit conversion constant = 28317 cc/SCFM x 1440 min/day x 1 mL/gm ÷3785 mL/g

NOTE: Radio-iodine isotopes I-133 and I-135 decay to form meta-stable and ground state radioactive daughter xenon isotopes, Xe-135m and Xe-133, and Xe-135m and Xe-135, respectively. During periods of primary-to-secondary leak rate, these radio-iodine isotopes build up in the secondary system (SG bulk water, SG crevices, blowdown demineralizers, and/or condensate polishers) and decay to contribute to the release of additional xenon activity.

B.2.2 Impact of Iodine Decay on Condenser Off Gas (COG) Count Rate

The impact of iodine decay on the total COG radiation monitor count rate, CR_{COG}, may be accounted for using the following equations:

$$\begin{aligned} CR_{COG} &\approx CR_1 + CR_2 + CR_3 + \ldots + CR_n + BKG \\ &\approx \left(\Sigma CR_j + BKG\right) \\ &\approx \left[\Sigma\left(A_{COG_j}\varepsilon_j\right)\right] + BKG \\ &\approx \frac{\left[\Sigma\left(\left[A_{RCS_j} + \dfrac{A_{RCS_i} f_j \lambda_j}{\lambda_i + \beta_{DIS}} e^{-\lambda_j T}\right]\varepsilon_j\right)\right] LR}{10773 F_{COG}} + BKG \end{aligned}$$

Equation B-4

Where:

LR	= Leak Rate (gpd)
BKG	= COG radiation monitor background (cpm)
$A_{COG,j}$	= Calculated or actual COG activity of noble gas radionuclide, j, due to primary to-secondary leak rate (μCi/cc)
ε_j	= Radiation monitor efficiency for gaseous radionuclide, j
$A_{RCS,j}$	= Measured RCS activity of noble gas radionuclide, j (μCi/g)
$A_{RCS,i}$	= Measured RCS activity of I-133 and I-135 isotope activities when decay significantly contributes to the COG Xe-135m, Xe-133, Xe-135m and/or Xe-135 component activities (μCi/g)
f_i	= Beta decay branching fraction of parent iodine isotope into meta-stable or ground state daughter xenon isotope
β_{DIS}	= Steam generator blowdown discharge constant ≈ blowdown mass-volume discharge rate ÷ SG mass-volume, (sec^{-1})
λ_j	= Decay constant for noble gas radioisotope, j (sec^{-1})
λ_i	= Decay constant for I-133 and I-135 isotopes (sec^{-1})
T	= Time between sampling and counting plus COG transient time (sec) (Because a transient decay time also exists for RCS sample lines, it may be acceptable to determine time, T, and neglect the RCS sample line and COG transient times.)
F_{COG}	= Total COG flow rate for evaluation period (cfm)
10773	= unit conversion constant = 28317 cc/cf x 1440 min/day x 1 mL/gm ÷3785 mL/g

A COG radiation monitor count rate to leak rate correlation factor, *CF,* can be calculated by substituting a primary-to-secondary leak rate of 1 gpd into Equation B-3, converting cpm to gpd.

$$CF \approx \left(CR_{COG} - BKG\right)$$

$$CF \approx \frac{\left[\Sigma\left(\left[A_{RCS_j} + \frac{A_{RCSi} f_j \lambda_j}{\lambda_i + \beta_{DIS}}\right]\varepsilon_j\right)\right]}{10773 F_{COG}}$$

Equation B-5

The primary-to-secondary leak rate may be estimated by dividing the correlation factor into the net (background subtracted) COG radiation monitor count rate:

$$LR = \frac{CR_{COG} - BKG}{CF}$$

Equation B-6

C EXAMPLE OF COMPUTER CALCULATED PRIMARY-TO-SECONDARY LEAK RATE FROM CONDENSER AIR EJECTOR

C.1 Overview

This appendix provides an example method for determining primary-to-secondary leak rate using a computer with continuous input from one or more air ejector (AE) process monitors. The monitors used in this method include:

- AE radiation monitor
- AE flow rate monitor

If a continuous flow rate input is not available, the method discusses manual input of a flow rate value into the computer that is updated periodically to reflect current conditions.

In addition to process monitor inputs, several constants are provided to the computer to convert the process monitor count rate and flow rate to gallons per day (gpd) at 77°F (25°C).

If this calculation is performed on a central plant process computer, control room operators have access to the continuous leak rate value for monitoring. Also, the resulting leak rates can be used to further calculate a rate of change.

C.2 Assumptions

The following conditions are needed for the calculation:

1. Computer communication with the AE radiation monitor
2. Computer communication with the AE flow rate monitor (optional)
3. Ability to update the computer with:
   - 3.1 AE radiation monitor background value (monitor reading with no leak rate) (cpm)
   - 3.2 AE radiation monitor total response factor (the summation of the product of the monitor isotopic efficiency and the current reactor coolant system (RCS) isotopic concentration)
   - 3.3 Conversion constant (units described below)
   - 3.4 A current flow rate value entered as a constant if continuous AE flow rate input is not available

C.3 Example AE Radiation Monitor Isotopic Efficiency Factors

Table C-1 shows example isotopic efficiency values for an AE radiation monitor provided by the monitor manufacturer. Each plant needs to obtain or determine these values for its installed monitor and the geometry it observes.

Table C-1
Example Isotope Efficency for an Air Ejector Radiation Monitor

Isotope	Detector Efficiency (CPM/μCi/cc)
Xe-133	3.38E7
Xe-135	1.11E8
Xe-137	1.39E8
Xe-138	1.14E8
Kr-85	1.06E8
Kr-85m	8.69E7
Kr-87	1.33E8
Kr-88	1.00E8
Kr-89	1.23E8
Ar-41	1.17E8
C-11	1E8

Detector efficiency values for all the gaseous isotopes observed in the Reactor Coolant System (RCS) may not be available from the equipment manufacturer. Detector efficiencies for isotopes not available from the manufacturer can be determined by plotting average beta energy (for a beta scintillator detector) versus detector efficiency for the isotopes available from the manufacturer. Then, using known or calculated average beta energies for the other isotopes, efficiencies for the other isotopes can be determined from the curve. See Figure C-1 at the end of this appendix for an example curve.

C.3.1 Calculation

The leak rate is calculated from the following equation:

Equation C-1

$$LR = 1.08 \times 10^4 \frac{(CR_{AE} - BKG)F_{AE}}{R}$$

Where:

LR = Primary-to-secondary leak rate (gpd corrected to 77°F)

1.08E4 = $\dfrac{\text{(28,317 cc air/cf air) x (24 hours/day) x (60 minutes/hour)}}{\text{(3785 g RCS/gallon RCS)}}$

CR_{AE} = AE radiation monitor reading (cpm)

BKG = AE radiation monitor background reading (cpm)

F_{AE} = AE flow rate (cfm at 77°F)

R = $\Sigma C_i k_i$ (total response factor)

Where:

C_i = Concentration of i^{th} isotope (gaseous) in RCS (μCi/g corrected for delay time if required)

k_i = AE radiation monitor isotopic detector efficiency for i^{th} isotope (cpm/μCi/cc)

C.3.2 Calculation Steps

Section C.3.2 provides a detailed calculation example.

Step 1. Determine the AE radiation monitor total response factor (R) and enter into the computer:

Table C-2
Example Isotope Concentration and Efficency Calculation

Isotope in RCS	RCS Activity (A_{RCS}) (μCi/g)	AE Radiation Monitor Isotopic Efficiency (k_i)	C_i x k_i
Ar-41	2.56E-03	1.17E+08	3.00E+05
Kr-85m	1.64E-04	8.69E+07	1.43E+04
Kr-87	3.06E-04	1.33E+08	4.07E+04
Kr-88	4.43E-04	1.00E+08	4.43E+04
Xe-133	8.48E-04	3.38E+07	2.87E+04
Xe-135	1.29E-03	1.11E+08	1.43E+05
Xe-138	1.79E-03	1.14E+08	2.04E+05

Total Response Factor ($\Sigma C_i k_i$) = R = 7.75E+05 cpm-cc air/g RCS

Step 2. Enter the AE radiation monitor background value into the computer.

Step 3. Enter the conversion constant of 1.08E4 into the computer.

Step 4. If necessary, enter the current AE flow rate into the computer.

Step 5. Example Calculation:

$$LR = 1.08 \times 10^4 \frac{(CR_{AE} - BKG)\,F_{AE}}{R}$$ **Equation C-2**

Where:

1.08E4	= A constant provided to the computer
CR_{AE}	= AE radiation monitor reading as read by the computer (cpm) As an example, use 250 cpm.
BKG	= AE radiation monitor background reading (cpm) As an example, use 100 cpm (constant entered to the computer).
F_{AE}	= AE flow rate as read by the computer (or provided as a constant if no input is given to the computer or if the monitor is inoperable) (cfm) As an example, use 7 cfm.
R	= 7.75E5 cpm-cc air/g RCS (a constant entered to the computer)

A monitor reading of 250 cpm, with an air ejector flow rate of 7 cfm, and a monitor background reading of 100 cpm would result in a computer calculated leak rate value of:

$$LR = 1.08 \times 10^4 \frac{(250\ cpm - 100\ cpm) \times 7\ cfm}{7.75E5 \dfrac{cpm \cdot cc\ air}{g\ RCS}} = 14.6\ gpd\ at\ 77°F$$ **Equation C-3**

C.4 Isotopic Considerations

It is important to ensure that the total response factor, *R*, incorporates all of the isotopes present in the AE grab sample. One utility has determined that the observed 511 keV peak in the grab sample analysis was due to the presence of C-11. This utility's experience is discussed below:

- In a past correlation of grab sample results to the condenser off gas reading, there was a significant difference. At the time of the sample, there was a low inventory of noble gas fission products due to low failed fuel. The nuclide providing the 511 keV peak appeared to contribute significantly to the monitor reading. Subsequent investigation of the peak attributed the peak to C-11 in the off gas sample.
- In the reactor coolant sample, the 511 keV peak is generally attributed to F-18. A series of counts on reactor coolant samples showed that part of the 511 keV peak could be attributed to C-11, even though the C-11 concentration accounted for less than 10% of the peak in the reactor coolant. C-11 is produced by a proton, neutron reaction with B-11. Therefore, estimates of the C-11 in reactor coolant can be made with a direct relationship between reactor power and boron.

- The significance of the peak in the off gas sample can be large during periods of low fuel failure. At this utility, the contribution ranges from approximately 25% at end of core life (low boron) to 97% at the beginning of core life (high boron). Therefore, the utility now calculates C-11 from the daily reactor coolant boron and power level and includes it in the total response factor used for the primary-to-secondary leak rate calculation.

C.5 Conclusions

The equation given in Section C.3.1 of this appendix will provide a continuous determination of primary-to-secondary leak rate based upon readings from continuous AE radiation monitor and flow rate instrumentation. Periodic updating of the computer with current RCS isotopic concentration and detector response factors, as well as periodic evaluation of appropriate monitor background values, is necessary.

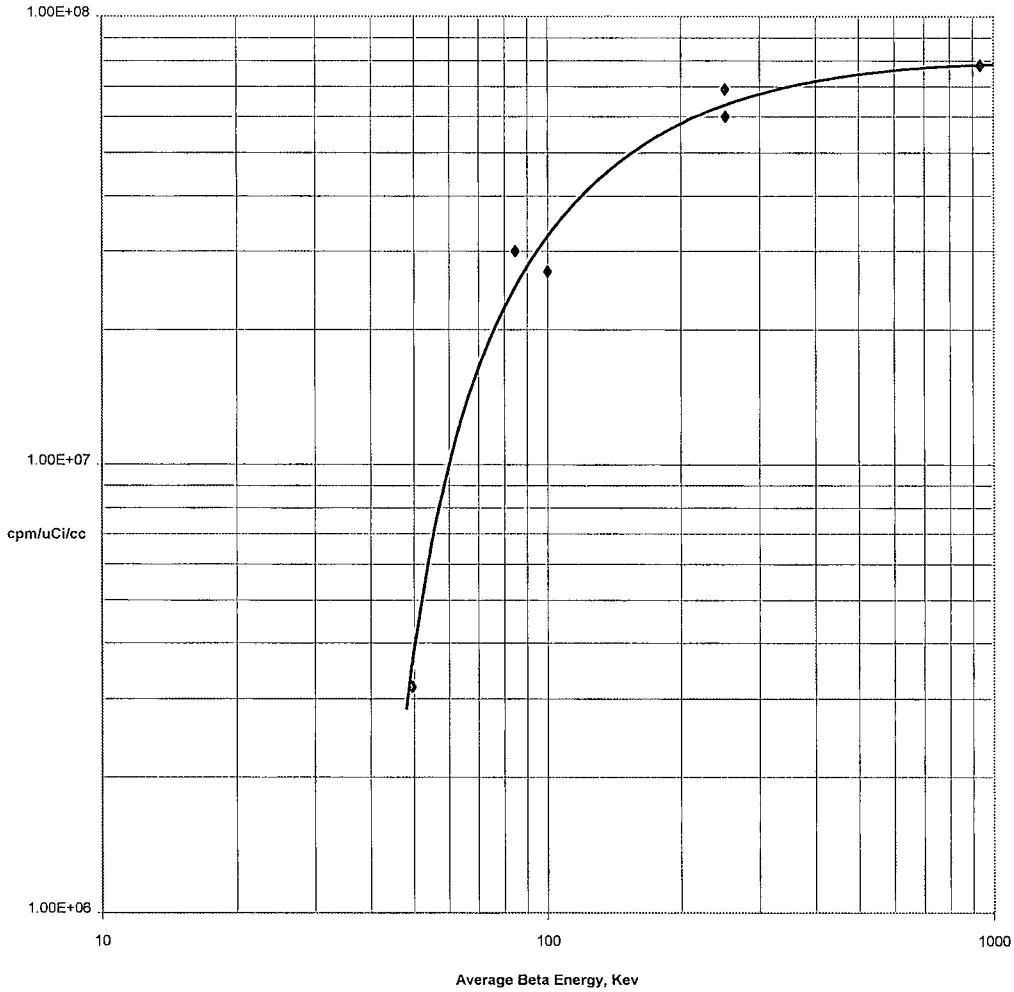

Figure C-1
Example Air Ejector Radiation Monitor Response Curve (Note That This is Not the Same Detector as in Table C-1)

D
PRIMARY-TO-SECONDARY LEAK QUANTIFICATION DURING NON-OPERATING CONDITIONS

D.1 Need for Determining Leak Rate During Shutdown Periods

As noted in Section 3, secondary-to-primary leakage can occur during shutdown periods. This section presents an evaluation methodology for leak rate determination when steam generator mass is not constant and blowdown may be in service. This is a more complex methodology than that presented in Section 5.6, since it addresses broader application.

D.2 Leak Rate Monitoring Prior to Power Operation

D.2.1 Plant Conditions

Four plant conditions (other than power operation) exist during which leak rate characterization by chemical or radionuclide measurements is desired:

- Mode 5, Cold Shutdown, after refueling or after shutdown
- Mode 4, Hot Shutdown, either during startup or shutdown conditions
- Mode 3, Hot Standby, either during startup or shutdown conditions
- Mode 2, Startup

Two operating progressions exist for the four conditions: the cited modes during a shutdown and those during a startup. Two common features exist: short-lived radio-nuclides are either absent or not being produced (such as during a shutdown where they may be present), and various plant systems are either isolated or operated infrequently. This limits leak rate monitoring using chemical and radionuclide measurements to the steam generators since the air ejector(s) will not be in service and steaming will only occur if a shutdown is in progress. Another consideration is that conditions may not be stable such as the blowdown flow rate and steam generator water level. These will impact the methodology used to quantify leak rate during non-operating modes.

D.2.2 Chemistry and Radionuclide Conditions

To use either a chemical or radionuclide tracer, a source must be characterized and the leaking system must be characterized by the same entity. This limits possible candidates to the following, with estimated levels:

Table D-1
Candidate Analytes for Determining Leak Rate During Non-Operating Conditions

Chemical Monitor	Reactor Coolant System Range	Radionuclide Monitor	Range
Boron	Several hundred ppm to refueling levels, 2000-2200 ppm	H-3	LLD to approximately 1 μCi/ml
Lithium	0 to approximately 6 ppm	Cs-134, Cs-137	LLD to approximately 10^{-2} μCi/ml

Prior to using some long-lived radionuclides plant staff should understand that there are issues that must be considered and understood prior to use for leak rate calculations. For example cobalt is not recommended for use as a leak rate indicator on plant start-up due to its fluctuating concentration in the primary circuit. Likewise, if the plant is being shut down, boron and lithium cannot be used since these species can return from hideout and give a false indication of current leak rate. The following are potential problems to be considered when selecting a chemical or radionuclide tracer:

- Xe-133 and Xe-135: These isotopes exist in the gas phase and cannot be quantified accurately using liquid samples.
- Co-58 and Co-60: Cobalt concentrations will change rapidly due to solubility changes and possible hideout/ hideout return phenomena.
- Cs-134 and Cs-137: These very soluble radioisotopes have been found to be entrained in secondary system magnetite surfaces and then return to the bulk water at varying rates long after past primary-to-secondary leak rate.
- Boron and Lithium: Concentrations of these chemical species can be significantly influenced by steam generator hideout/ hideout return mass transport during start-up and shutdown periods.

D.2.3 General Equation for Leak Rate Monitoring in the Steam Generator

The basic equation for leak rate involving a retention process (e.g., the release from the steam generator is not instantaneous) for a radionuclide with a constant source term is:

$$M \bullet \frac{dA}{dt} = L \bullet A_{RCS} - \lambda MA - \delta MA - Q_R \bullet A$$

$$A_t = \frac{\frac{L}{M} \bullet A_{RCS}}{\left(\lambda + \delta + \frac{Q_R}{M}\right)} \left[1 - e^{-\left(\lambda+\delta+\frac{Q_R}{M}\right)\bullet t}\right] + A_0 \bullet e^{-\left(\lambda+\delta+\frac{Q_R}{M}\right)\bullet t} \qquad \textbf{Equation D-1}$$

Where:

A_t	= activity in the secondary system, µCi/g (A_o at time =0)
L	= leak rate, g/sec
M	= mass of liquid water in steam generator, g
A_{RCS}	= activity in the reactor coolant, µCi/g
λ	= radioactive decay constant, sec^{-1}
δ	= constant to account for adsorption/hideout on plant surfaces, sec^{-1} (zero for H-3)
Q_R	= physical removal term such as blowdown or leak rate, g/sec
t	= time, sec

The mass balance equation neglects steam carryover, chemical reactions, and return via the feedwater. Steam carryover and return via the feedwater are not considered for non-operating conditions. The time-dependent equation may be simplified as follows:

$$A_t = \frac{L \bullet A_{RCS}}{M\alpha}\left[1 - e^{-\alpha\bullet t}\right] + A_0 \bullet e^{-\alpha\bullet t} \qquad \textbf{Equation D-2}$$

where α is the net removal constant for all removal processes from the steam generator. The time-dependent equation also applies for chemical species (i.e., $\lambda = 0$, C_t and C_{RCS} is the species):

$$C_t = \frac{L \bullet C_{RCS}}{M\alpha}\left[1 - e^{-\alpha\bullet t}\right] + C_0 \bullet e^{-\alpha\bullet t} \qquad \textbf{Equation D-3}$$

In the forms given above the time that the leak started and the initial concentration must be known to determine the leakage characteristics. However, this problem is easily overcome by recognizing that the equations above are valid between any two time points, as follows:

$$A_2 = \frac{L \bullet A_{RCS}}{M\alpha}\left[1 - e^{-\alpha \bullet \Delta t}\right] + A_1 \bullet e^{-\alpha \bullet \Delta t}$$ **Equation D-4**

as long as the other parameters (L, A_{RCS}, M, and α) are constant during the interval. Rearranging the equation above gives the following expression for the leakage, L:

$$L = \frac{A_2 - A_1 \bullet e^{-\alpha \bullet \Delta t}}{\left[1 - e^{-\alpha \bullet \Delta t}\right]} \frac{M\alpha}{A_{RCS}}$$ **Equation D-5**

Equation D-5 yields leakage in the familiar units of gallons per day based on room temperature where the steam generator water mass M is calculated using the volume of water and room-temperature density, the times are in days, and the net removal constant α is in day^{-1}.

If, as is often the case, the other parameters in the above equation (e.g., M, the water mass in the SG) are not constant, it is still possible to approximate the leakage using the above method by approximating the changing parameters as piecewise constant. A_t, the measured secondary activity at time t, can be calculated by solving Equation D-5 in a piecewise manner. Specifically, the full time interval of interest (i.e., from time t_0 to time t) may be evaluated as two or more successive intermediate intervals during which the values of L, A_{RCS}, M, and α were constant. For example, the secondary activity at the end of the first intermediate interval, A_2, is determined using A_{RCS1}, M_1, and α_1, which are constant over the time interval from t_1 to t_2. A_3 can then be calculated using the calculated value of A_2 and the values of A_{RCS2}, M_2 and α_2, valid over the time interval from t_2 to t_3, and so forth for any number of intervals. The leakage can then be determined such that the final calculated secondary side activity matches the measured activity at time, t. Explicit equations describing the application of this method are not provided herein as this method is most conveniently applied using a spreadsheet.

Reference [1] presents an evaluation of tritium diffusion and the significance of this tritium source term to the total secondary system tritium concentration. It is acknowledged that tritium diffusion is a known source of secondary tritium activity, but it was determined that it is a minor contributor relative to primary-to-secondary leak rate and that it should not be included in the tritium calculational methodology.

D.2.4 Considerations for Calculating Leak Rate During Non-Operating Conditions

D.2.4.1 The Net Removal Constant α

The net removal constant α has three components for non-steaming conditions:

$$\alpha = \lambda + \delta + \frac{Q_R}{M}$$ **Equation D-6**

For chemical species $\lambda = 0$, and for the candidate radionuclides H-3, Cs-134, and Cs-137, $\lambda \approx 0$. Thus,

$$\alpha \cong \delta + \frac{Q_R}{M}$$ **Equation D-7**

One problem with the above is that δ, the term to account for adsorption/hideout on plant surfaces, is not known and cannot be quantified, except for tritium whose $\delta = 0$. The only practical solution is to control the blowdown so that blowdown becomes the major removal mechanism. Adsorption/hideout is expected to be minor for Modes 5 and 4, but could be significant at greater than 350°F during Modes 3 and 2.

D.2.4.2 Practical Consideration with Steam Generator Parameters and Sampling

Another consideration is that steam generator water levels may change as plant conditions change. Although one can set up equations to account for each level change, this may not be practical. A more practical concern is the mass of water in the steam generator during non-operating conditions. The following is used in the EPRI Hideout Return Spreadsheet for determining the amount of water in various steam generators. The mass calculated from this equation is the total water mass in the steam generator. Below a few percent power the actual mass in the recirculation or mixing path may be significantly different from this value depending on the mixing methodology.

$$SG_{Mass\,of\,water\,(lbm)} = \left(\left(V_a\,x\,\%NR + V_b\right)x\left(\left(P_a\,x\,\%NR\,x\,\%Power\right)^2 + \left(P_b\,x\,\%Power\right) + P_c\right)\right)x\,\rho$$

Equation D-8

Where:

V_a/V_b = Constant based on Volume

$P_a/P_b/P_c$ = Fitting Constants (Constant)

Table D-2
SG Volume by Vendor and Model

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A major concern is sampling the steam generator during blowdown isolation or shutdown conditions. If the plant is at temperature during blowdown isolation, blowdown samples may not be representative as a result of excessive sample transport times. Sample collection time should reflect sample transport time in the sample line if delays are significant. During shutdown periods, with or without blowdown isolation, pressure may not be available to provide sufficient head for sample flow; local samples must be withdrawn for analysis. This is a particular concern during layup conditions. If a recirculation pump is not used, local samples must be collected, and purging requirements may not be defined. Chemicals and radionuclides in the reactor coolant system introduced into a steam generator during non-operating conditions will not be mixed unless temperature is sufficient to achieve thermal mixing, or physical mixing (e.g., nitrogen sparging or a recirculation skid) is used. Replicate samples should be collected at the highest practicable purge rate to establish sample validity.

Another consideration in some steam generator designs is the effect of sample location in the steam generator. Some steam generators have provisions for sampling the downcomer region and the blowdown, and impurity concentrations are different at these locations. High blowdown flow rates reduce impurity levels, but in some designs higher blowdown flow rates direct more feedwater toward the blowdown lane, and the lower impurities are a result of analyzing samples enriched with feedwater rather than the water in contact with the steam generator tubes. Special considerations should be given to sampling the blowdown of recirculating steam generators where the feedwater is injected in a manner that directs the incoming feedwater to the path of the blowdown extraction piping, particularly during shutdown periods. The blowdown samples withdrawn during periods of feedwater injection will be enriched with feedwater.

The current recommendation is to ensure that the contents of the steam generator are mixed to the degree practicable during shutdown conditions prior to withdrawing a sample for analysis. Sampling biases during these conditions are expected to have a major impact on leak rate evaluations. Having limited blowdown is also desirable to ensure that the removal term α is controlled by a known quantity (blowdown) that is thought to be greater than δ, a non-quantifiable term that accounts for adsorption/hideout on plant surfaces.

D.2.5 Estimated Steam Generator Concentrations under Different Conditions

D.2.5.1 Case 1

Steam generator boron, lithium, tritium, and cesium levels were calculated for a Westinghouse Model 51 steam generator assuming the level to 50% narrow range at ambient conditions; reactor coolant concentrations were assumed to be:

Table D-3
Normal Concentrations for Consideration in Steam Generator

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Except for tritium, the removal term α was assumed to be:

$$\alpha = \delta + \frac{Q_R}{M} \qquad \textbf{Equation D-9}$$

$$= 1 \times 10^{-2} + \frac{10 \text{ gal/min} \times 60 \text{ min/hr}}{27{,}360 \text{ gal}}$$

$$= 3.193 \times 10^{-2} \text{ hr}^{-1}$$

If a value had not been assumed for α, the levels in the steam generator eventually would approach reactor coolant levels and would build up in a linear manner with time. The calculated steam generator concentrations as a function of time are plotted in Figures D-1 through D-4. The calculated steam generator levels and the figures indicate the following relative to determining leak rate during shutdown conditions.

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Figure D-1
Calculated Secondary Side Steam Generator Boron Levels vs. Time (Hours) for Different Leaks

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Figure D-2
Calculated Secondary Side Steam Generator Lithium Levels vs. Time (Hours) for Different Leaks

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Figure D-3
Calculated Secondary Side Steam Generator Tritium Levels vs. Time (Hours) for Different Leaks

Content deleted - EPRI Proprietary

Figure D-4
Calculated Secondary Side Steam Generator Cesium Levels vs. Time (Hours) for Different Leaks

<u>Boron</u>

The mannitol titration method has been used to determine boron down to approximately 1 ppm. This method may not be adequately sensitive, and alternate analysis techniques such as using an IEC (ion exclusion chromatography) column with heptafluorobutyric acid eluent and 5 mM

TBAOH (tetrabutylammonium hydroxide) regenerate should be considered (boron between 0.5 and 50 ppm can be measured with reasonable accuracy by this method). The curcumin colorimetric method (ASTM D 3082 - 92) is valid from 0.1 to 1.0 ppm, and direct-current argon plasma atomic emission spectroscopy possibly could be used down as low as 50 ppb. The calculations indicate that leak rate could be quantified with the mannitol titration method if the reactor coolant boron were at refueling concentrations (e.g., 1800 ppm or greater) and the leak rate were 30 gpd or greater. Other methods with increased sensitivity would be required at lower leak rate or if the reactor coolant boron were at the hundred ppm level. Confusion will result if the plant is on boric acid treatment (BAT) or if boron hideout return is occurring. The overall conclusion is that boron should be used for gross indications of leak rate if levels are increasing under stable steam generator conditions.

Lithium

Lithium levels in Figure D-2 indicate that the common AA method lacks adequate sensitivity to quantify lithium in a leaking steam generator under non-operating conditions. IC techniques, however, can be used to quantify lithium at the 20 ppt level. Direct-current argon plasma atomic emission spectroscopy possibly could be used, but the quantification level has not been reported. Lithium also will be in hideout return from previous operation. The overall conclusion is that lithium should be used for gross indications of leak rate if levels are increasing under stable steam generator conditions.

Tritium

Reactor coolant tritium levels of 0.5 and 0.05 μCi/ml were used to calculate steam generator tritium levels at different leak rate shown in Figure D-3. The usual minimum sensitivity for Liquid Scintillation Counting (LSC) counting techniques is 2-5 x 10^{-6} μCi/ml. The calculations indicate that tritium can be used to quantify leak rate under non-operating conditions, provided that the reactor coolant tritium is greater than approximately 0.05 μCi/ml, although lower levels may be valid. The main limitation will be the initial tritium level in the steam generator water prior to the current leak rate; levels initially greater than approximately 10^{-5} μCi/ml will limit the sensitivity. Adsorption/hideout is not an issue with tritium (e.g., $\delta = 0$).

Long-Lived Cesium Activity

Figure D-4 shows the calculated Cs-134/Cs-137 activity in the steam generator water at different leak rate levels with 5 x 10^{-3} μCi/ml and 1 x 10^{-4} μCi/ml in the reactor coolant. Levels less than approximately 10^{-5} μCi/ml can be quantified by gamma spectrometry, but the calculations indicate that long-lived cesium activity cannot be used to quantify leak rate. Cesium also will behave chemically similarly to sodium and hideout return from previous operation may mask cesium from current leak rate.

D.2.5.2 Case 2

Plant experience at several units has confirmed that nuisance primary-to-secondary leak rate, as enhanced by fuel failures, can occur from manufacturing defects not related to steam generator tube degradation. This situation then results in low-level gamma activity in the plant's secondary system. The situation can be further complicated in multi-unit sites where some secondary system waters are shared between units.

Figures D-5 through D-7 illustrate such an occurrence at a US PWR. The scenario experienced at the site was marked by the following events:

- Gamma spectral analysis of secondary system sample panel cation columns indicated an increase in primary-to-secondary leak rate from the Unit 2 SG2 (Figure D-5).
- The increase was the result of a new fuel defect and not steam generator tube degradation, as was confirmed by Na-24 trends (Figure D-7).
- The increase in primary-to-secondary leak rate was determined to be due to pre-existing steam generator manufacturing defects, illuminated by the new fuel defect.

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Figure D-5
Unit 2 SG2 Primary-to-Secondary Leak Rate From CC Cs-137 Data

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Figure D-6
Unit 2 RCS and SG Cesium-137

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Figure D-7
Unit 2 Steam Generator Sodium-24 Concentrations

D.3 Conclusions

This appendix provides the calculational methodology to determine leak rate during non-operating conditions. Leak rate can be quantified with tritium in the steam generator, provided that the reactor coolant level is greater than approximately 0.05 μCi/ml and the initial level in the steam generator prior to the current leak rate is less than approximately 10^{-5} μCi/ml. Boron and lithium can be used for a gross indication of leak rate, and under limited conditions these can be used to quantify leak rate. Long-lived cesium activity cannot be used for quantification.

A major consideration for characterizing leak rate during non-operating conditions is steam generator sample validity. Special controls may be necessary to ensure sample validity, regardless of the method used. Quantifying leak rate with chemical tracers has an additional limitation in that hideout return/hideout/adsorption will confuse and limit the interpretation. This is not an issue with tritium. Steam generator water levels should be maintained relatively constant to increase accuracy. To limit the unknown effect of adsorption/hideout of boron, lithium, and cesium, blowdown is recommended. The blowdown flow rate may require a temperature correction to account for differences in the calibration conditions and the non-operating conditions, depending on the plant. Blowdown is not necessary for tritium since adsorption/hideout is not an issue.

D.4 References

1. *Tritium Diffusion Assessments*, Dominion Engineering, Inc., Reston, VA: 2009. M-5593-00-04, Revision 3.

E
ARGON INJECTION

E.1 Purpose

This appendix describes the use of argon gas in the reactor coolant system (RCS) to improve the sensitivity of radiation detection instrumentation to the presence of primary-to-secondary leak rate.

E.2 Background

The required sensitivity of primary-to-secondary leak rate (PSL) monitoring has evolved over recent years. Power plant personnel have had to improve both process instrumentation and grab sampling and analysis techniques to achieve this required sensitivity.

These *Guidelines* require that radiation monitoring equipment be able to detect 30 gpd total leak rate, which can be especially challenging for plants without main steam line N-16 monitors and for those operating without fuel defects. These plants rely primarily upon condenser gas removal radiation monitors for process monitoring.

In the absence of fuel defects, some plants find that the presence of Ar-41 significantly improves the sensitivity of leak rate calculations. In some cases, the Ar-41 activity is relatively high (1E-3 – 1E-2 μCi/cc range) at the beginning of a fuel cycle. This is due to the refueling water becoming saturated with air, which is approximately 0.9% argon. During startup, oxygen is removed (sometimes chemically, using hydrazine), leaving behind the argon in the RCS. The inventory of Ar-40 in the RCS is activated over time [Ar-40 (n,γ) Ar-41].

However, during the fuel cycle, the inventory of Ar-40 is consumed, and the concentration of Ar-41 eventually decreases. This results in the degradation of sensitivity for units which rely on Ar-41 for leak rate quantification. Some plants may find that make-up water provides a source of argon for activation in the RCS. Improvements in make-up water processing to remove dissolved gases (primarily oxygen) may reduce or eliminate this potential source of argon. Therefore, reliance upon the Ar-41 concentration in the RCS is not always dependable.

As plants have improved refueling processes, activities such as vacuum refill of the RCS have resulted in the removal of oxygen, as well as other gases, like argon. The Ar-41 concentration at the beginning of a fuel cycle is significantly lower than observed in the past cycles. This presents an additional challenge because the once beneficial Ar-41 (at least in the first part of a fuel cycle) is no longer available as a source term to improve primary-to-secondary leak rate detection capabilities.

To make RCS Ar-41 a more dependable isotope for primary-to-secondary leak rate monitoring, some plants have now implemented the injection of natural argon gas into the RCS to maintain the concentration of Ar-41 at a consistent level. The concentration is maintained at sufficiently high level to ensure that process instrumentation is sensitive to 30 gpd, and that grab sample analysis is sensitive to 5 gpd. The concentration of Ar-41 that achieves this sensitivity has been found to be near the concentration range previously observed in the RCS at the beginning of a fuel cycle, before improvements in operation practices reduced the argon inventory.

Argon injection can be relatively simple to implement, inexpensive to maintain, and easy to control. It is recommended that any addition of natural argon gas to the RCS be performed using a process specifically designed for this purpose. Techniques such as adding air or argon to make-up water are not recommended. Such techniques would not provide targeted and timely responses to RCS Ar-41 changes.

E.3 Properties of Argon

Tables Table E-1 and Table E-2 give properties of natural argon and Ar-41, respectively.

Table E-1
Isotopic Properties of Argon

Natural Argon		
Natural isotopic ratios	Ar-36	0.3365%
	Ar-38	0.0632%
	Ar-40	99.6003%
Concentration in atmosphere		0.934%
Ar-40 thermal neutron activation cross section		0.66 Barns
Other Considerations – Inert, readily available, and inexpensive		

Table E-2
Basic Radioactive Properties

Argon-41		
Half-life		1.83 hours
Decay energies	1293.60 keV gamma	99.20% of disintegrations
	1198.30 keV beta	99.17% of disintegrations
Other Considerations	Unaffected by fuel defects	Ar-41 familiar to plant personnel

E.4 Argon Gas Volume Needed to Maintain RCS Ar-41

Utilities currently injecting natural argon into the RCS have found that an RCS Ar-41 activity maintained at approximately 0.1 μCi/cc provides condenser gas removal system radiation monitor sensitivity to primary-to-secondary leak rate well below the 30 gpd requirement.

E.4.1 Concentration of Natural Argon

Equation E-1 can be used to determine the concentration of natural argon required to maintain a given concentration of Ar-41.

$$^{nat}Ar_{RCS} \approx \frac{^{41}Ar_{RCS}}{f_{^{40}Ar} \times \delta_{th} \times K \times eff\Phi_{th}}$$

Equation E-1

Where:

$^{nat}Ar_{RCS}$ = concentration of argon gas in the RCS (cc/kg)

$^{41}Ar_{RCS}$ = concentration of Ar-41 in the RCS (μCi/cc)

δ_{th} = thermal neutron cross section for Ar-41 (6.6E-25 cm^2)

$f_{^{40}Ar}$ = Ar-40 fraction of natural argon gas (0.9960)

K = conversion factor

$$= \frac{1\ \text{mole}}{2.4E4\ \text{std cc}} \times \frac{6.023E23\text{atoms}}{\text{mole}} \times \frac{1\ \mu Ci}{3.7E4\ \text{disintegrations/sec}} \times \frac{1\ \text{kg}}{1000\ \text{gm}}$$

= 6.78E11 atom/cc x μCi/dps x kg/gm

Where:

$eff\Phi_{th}$ = effective thermal neutron flux at 100% reactor power and can be calculated by one of two methods listed below:

1. Calculated from known average core thermal neutron flux

$$eff\Phi_{th} = f_{core} \times \Phi_{th}$$

Where:

f_{core} = fraction of RCS mass exposed to core neutron flux during power operation

$$\frac{\text{Core water volume}}{\text{Total RCS water volume}}$$

Φ_{th} = average core thermal neutron flux at 100% power (n/cm^2/sec)

2. Empirically determined from RCS lithium grow-in by $^{10}B(n,\alpha)^{7}Li$

$R = N \times \sigma \times eff\Phi_{th}$ then, rearranging:

$$eff\Phi_{th} = \frac{R}{N \times \sigma}$$

Where:

R = Production rate of ^{7}Li in the RCS (atom/second-gram)

N = Number of target atoms (boron) per gram of solution

σ = reaction cross-section (cm^2)

Example: 0.115 ppm/day ^{7}Li grows in at a RCS concentration of 1282 ppm B

$$R = \frac{0.115E-6\,g\ ^{7}Li}{g\ RCS-day} \times \frac{mole}{7.016\,g\ ^{7}Li} \times \frac{6.023E+23\ atoms}{mole} \times \frac{day}{8.64E+4\ sec}$$

$= 1.14E+11$ atoms/sec-gram RCS

$$N = \frac{1.282E-3\,g\ B}{g\ RCS} \times \frac{6.023E+23\ atom\ B}{10.811\,g\ B} \times \frac{0.199\ atoms\ ^{10}B}{atom\ B}$$

$= 1.42E+19$ atoms ^{10}B/g RCS

$$eff\Phi_{th} = \frac{1.14E+11\ atoms/sec-g\ RCS}{(1.42E+19\ atoms\ ^{10}B/g\ RCS)\times(3838E\text{-}24\ cm^2)}$$

$= 2.09E+12\ n/cm^2$-sec

For example using the inputs in Table E-3, yields the following:

$$^{nat}Ar_{RCS} = \frac{0.1\mu Ci/cc}{(0.9960)\times(0.05)\times(6.6E-25cm^2)(4E+13n/cm^2/sec)\times[6.78E11(atoms/cc)(\mu Ci/dps)(kg/gm)]}$$

$^{nat}Ar_{RCS} = 0.1$ cc/kg

Table E-3
Nominal Values for a Typical PWR

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E.4.2 Volume of Natural Argon

Equation E-2 can be used to estimate the total volume of natural argon required to achieve 0.1 cc/kg in the RCS. This equation includes VCT gas space argon volume. The equation does not include argon in the pressurizer and each site should review applicability before implementing. This value can be difficult to determine and can be very plant specific, depending upon how the pressurizer is controlled. Therefore, the equation may underestimate the volume of argon required to achieve the desired concentration.

$$ {}^{nat}Ar_{tvol} = \left[{}^{nat}Ar_{RCS}V_{RCS}\rho K^{'}\right] + \left[\left(\frac{{}^{nat}Ar_{RCS}\, x\, K^{''}}{K_H}\, x \frac{1}{1\,atm}\right) x\, VCT_g\right] $$ **Equation E-2**

Where:

${}^{nat}Ar_{tvol}$	= total volume of argon gas in the RCS (liters)
${}^{nat}Ar_{RCS}$	= concentration of natural argon gas in the RCS (cc/kg)
V_{RCS}	= total water volume in RCS (ft^3)
ρ	= specific gravity of RCS at t_{avg} (kg/L)
$K^{'}$	= conversion factor $\frac{28.317\ \mathrm{L}}{\mathrm{ft}^3} \times \frac{1\mathrm{L}}{1000\ \mathrm{cc}}$ = 2.8317E-02 L^2/(ft^3-cc)
$K^{''}$	= conversion factor $\frac{1\ \mathrm{mole}}{22.4\ \mathrm{L@STP}} \times \frac{1\ \mathrm{kg}}{1\ \mathrm{L}} \times \frac{1\ \mathrm{L}}{1000\ \mathrm{cc}} = 4.46\mathrm{E}-05 \frac{\mathrm{mole-kg-L}}{\mathrm{L}^2-\mathrm{cc}}$
VCT_g	= Volume control tank gas volume (liters)
k_H	= Henry's constant $\left(\frac{\mathrm{mole/L}}{\mathrm{atmosphere}}\right)$ for argon at 298 K (Actual VCT Temp.)

Example:

Table E-4
Nominal Values for a Typical PWR

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$$ {}^{nat}Ar_{tvol} = \left[0.1\frac{cc}{kg}\times 11{,}000\ ft^3\times 0.707\frac{kg}{L}\times 2.8317E-02\frac{L^2}{ft^3-cc}\right] + \left[\frac{0.1\frac{cc}{kg}\times 4.46E\text{-}05\frac{mole\text{-}kg\text{-}L}{L^2-cc}}{1.4E-03\frac{mole}{(L\text{-}atm)}}\times\frac{1}{1\,atm}\times 6500\ L\right] $$

$$ {}^{nat}Ar_{tvol} = 42.7\text{ L of natural argon gas} $$

E.5 Injection Techniques Currently Used In Industry

Multiple techniques may be employed to add argon to the RCS. Two successful techniques are described in the following sections. Both systems add argon directly into the volume control tank.

E.5.1 Low Flow Rate Injection Technique

One technique injects argon into the VCT liquid space. This technique can take advantage of an existing RCS zinc injection system. In the absence of an injection system specifically for zinc injection, a demineralized water injection system could also be used. The injection line can be added to the letdown heat exchanger flow going to the volume control tank.

The low flow rate injection system includes:

1. Pressurized argon gas cylinder (e.g., 225 ft^3 cylinder will last years)
2. Normal gas cylinder pressure regulator
3. Indicating gas mass flow controller (10 cc/min max is adequate)
4. Motive water (e.g., zinc injection) flow (0.5 gpm is adequate)

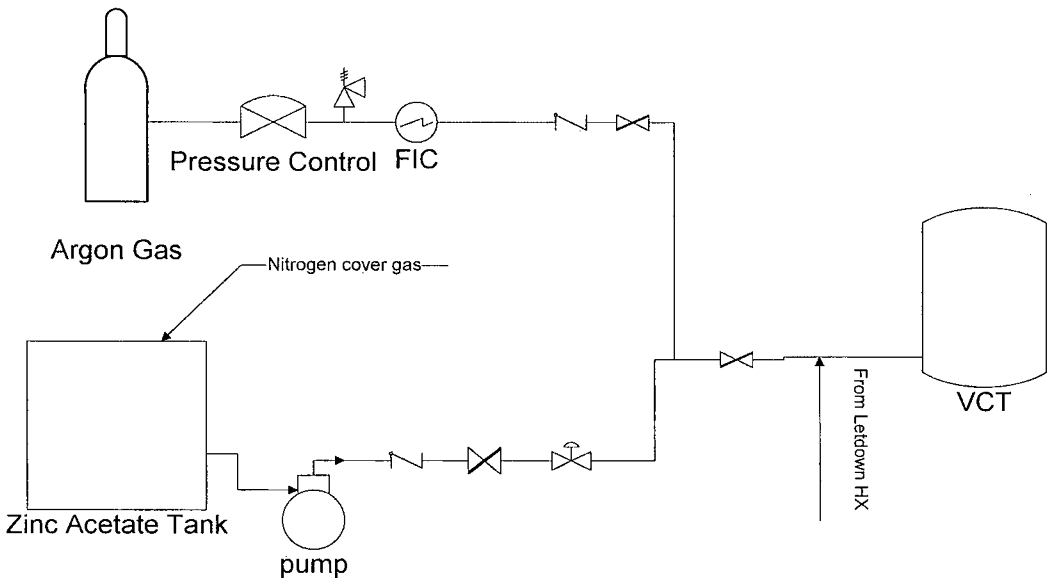

Figure E-1
Simplified Low Flow Rate Argon Injection System

E.5.2 High Flow Rate Injection Technique

The second example technique injects argon gas directly into the VCT gas space. This can be accomplished by injecting through the VCT gas space sample line. This technique most likely requires less physical change to existing plant equipment.

The high flow rate injection system includes:

1. Pressurized argon gas cylinder (e.g., 225 ft^3 cylinder will last years)
2. Normal gas cylinder pressure regulator
3. Gas flow rotameter
4. Quick connection to existing VCT gas space sample line

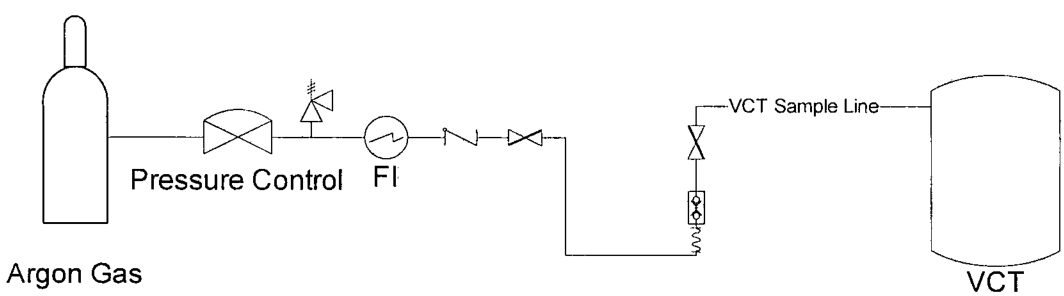

Figure E-2
High Flow Rate Argon Injection System

E.6 Natural Argon Gas Activation in the RCS

Ar-41 is essentially the only activation product of concern when natural argon gas is injected into the RCS. However, several activation products are created by the neutron activation of Ar-36, Ar-38 and Ar-40. The buildup of chloride, potassium, calcium, and sulfur in the RCS are insignificant.

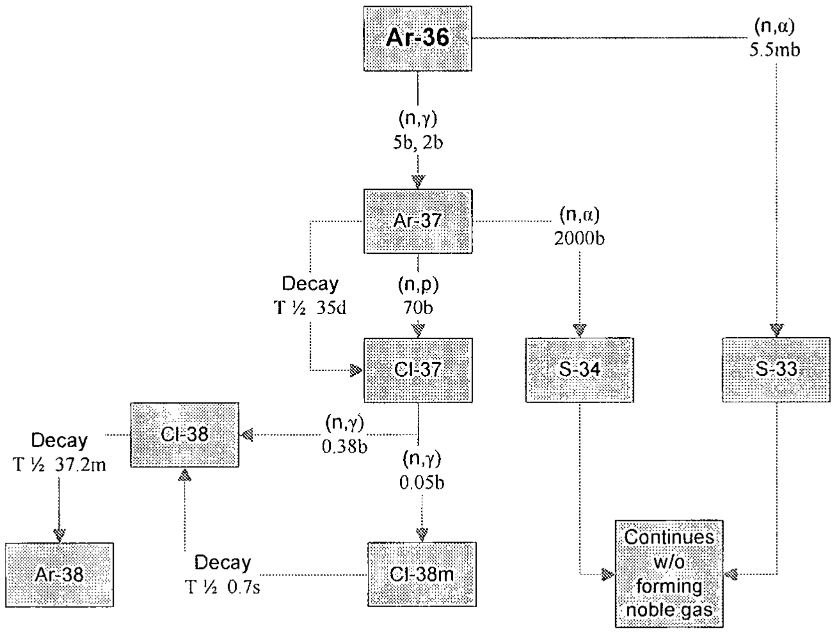

Figure E-3
Ar-36 Neutron Activation

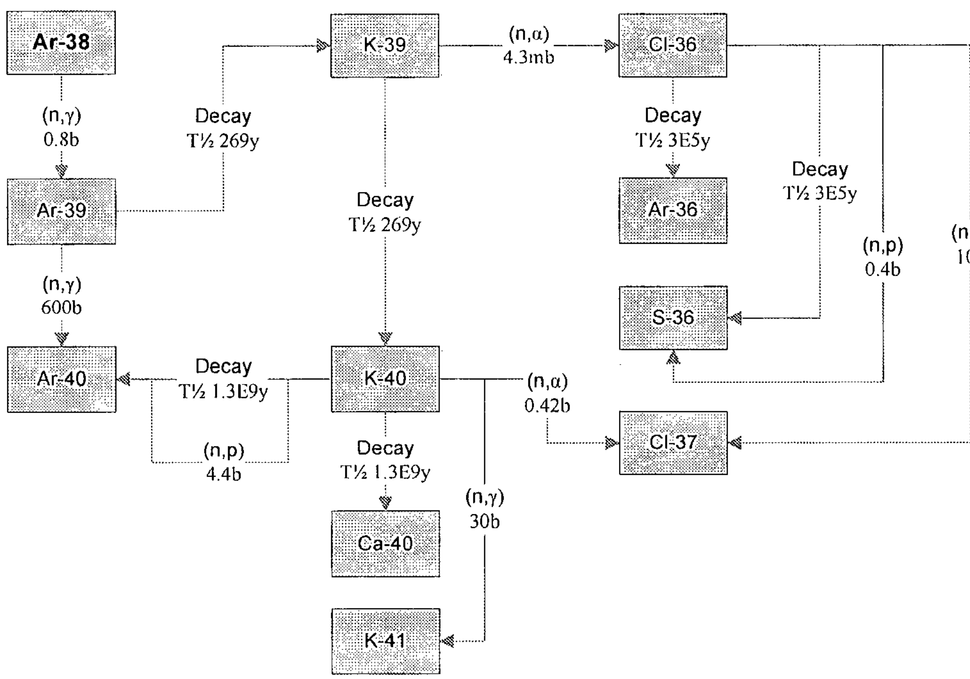

Figure E-4
Ar-38 Neutron Activation

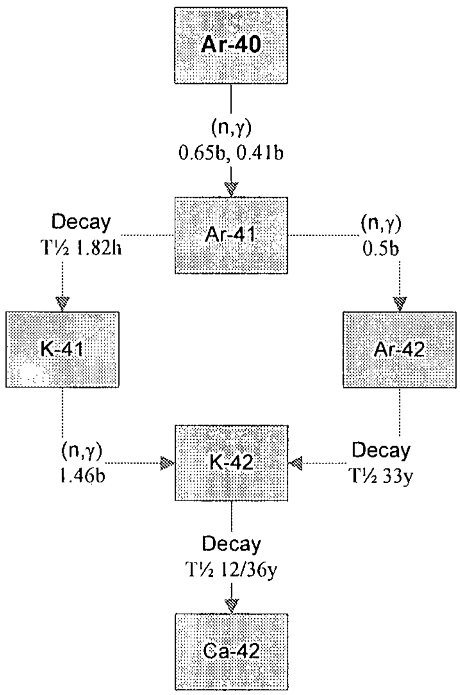

Figure E-5
Ar-40 Neutron Activation

E.7 Monitoring Recommendations

Plant maneuvers can result in significant changes in the Ar-41 concentration in the RCS. Plant personnel should implement routine practices to evaluate these changes.

E.7.1 Argon Injection System Monitoring

Chemistry personnel should establish an acceptable range of RCS Ar-41 concentration to maintain. A range successfully used at one plant is described below:

Table E-5
Example Ar-41 Trigger Points

[Ar-41] (µCi/cc)	Condition/Limit	Action
≥ 0.15	High Limit	Recommend VCT degas
≥ 0.10	Upper Target Limit	Verify injection secured
0.05 - 0.10	Normal Target Range	None
≤0.05	Lower Target Limit	Initiate injection

The following should be noted regarding the example Ar-41 concentration values shown in Table E-5:

1. The high limit value may be based upon injection system design change requirements or some other administrative limit.
2. The normal target range may be selected to avoid drastic changes in monitor response.
3. The lower target limit may be based upon the ability to detect 30 gpd.

While argon injection is occurring, personnel should establish routine practices to ensure expected system configuration is maintained.

E.7.1.1 Low Flow Rate Injection System

While physically injecting argon (sometimes for several days in a row), the following should be routinely performed/verified:

1. Injection flow rate is as expected
2. Gas supply is sufficient (e.g., gas cylinder pressure is adequate)
3. Increased RCS sampling to monitor Ar-41 concentration change

E.7.1.2 High Flow Rate Injection System

Injections at a flow rate of approximately 400 cc/min for approximately 12 minutes per week are normally adequate to maintain the RCS Ar-41 concentration within the desired range. While physically injecting argon at these high flow rates, the following recommended actions should be performed / verified:

- The entire evolution should be continually monitored by plant personnel.
- Ensure that a VCT gas space purge to the Waste Gas system is not in progress OR that a purge is not scheduled for at least 24 hours.
- Ensure that the expected injection flow rate is occurring.
- The control room should be notified prior to injection so that operations can monitor for any unusual conditions.
- Ensure that the gas supply is sufficient (e.g., gas cylinder pressure is adequate).
- Increase RCS sampling to monitor Ar-41 concentration change.

E.8 VCT Degas

Degassing of the VCT to the gaseous radioactive waste (GRW) system has two effects that should be evaluated.

E.8.1 Increase in GRW System Dose Rates

Operations personnel should notify radiation protection personnel prior to performing a VCT degas such that appropriate monitoring and personnel access can be evaluated. The following points are some issues that should be considered:

1. Dose rates will temporarily increase in the GRW between the VCT and the waste gas decay tanks.
2. In some cases, certain parts of the plant may become high radiation areas.
3. These changes are short lived, however, because of the relatively short half-life of Ar-41 (1.83 hours).

E.8.2 Decrease in RCS Ar-41

Operations personnel should notify chemistry personnel prior to performing a VCT degas so that appropriate evaluations can be anticipated and scheduled. These evaluations and scheduling may include the following:

1. Schedule sampling of the RCS to determine resulting Ar-41 concentration.
   a. This sample should not be taken immediately after the VCT degas because a new Ar-41 concentration equilibrium will not yet have been established.
   b. As an initial criterion, it is recommended that the sample(s) be taken within approximately 24 hours. Each plant can establish a more appropriate time period based upon experience.
2. Based upon resulting Ar-41 concentration, recalculate condenser gas removal system radiation monitor response factor.
3. Schedule argon injection to re-establish target Ar-41 concentration.

E.9 Reactor Power Changes

E.9.1 Reactor Power Decrease

If reactor power is decreased, the Ar-41 concentration will also decrease and establish a relatively steady concentration at a lower level. This results in a non-conservative change in condenser gas removal system radiation monitor response factor.

As an initial criterion, it is recommended that chemistry personnel perform the following after a power decrease of $\geq$10%:

1. Schedule sampling of the RCS to determine resulting Ar-41 concentration. A sample can be taken shortly after reaching the new power level (5-10 hours), because the new Ar-41 equilibrium is established after several half-lives have elapsed.
2. Based upon resulting Ar-41 concentration, recalculate condenser gas removal system radiation monitor response factor.

A different power decrease criterion may be established by each plant, based upon experience.

E.9.2 Reactor Power Increase

During periods of power ascension, the concentration of RCS Ar-41 will increase. During this time period, the condenser gas removal system radiation monitoring system response to primary-to-secondary leak rate will actually become more conservative, i.e., will overestimate leak rate. Therefore, it is not necessary, from a practical standpoint, to sample the RCS solely to recalculate the radiation monitor response factor. Since the monitor's response is more conservative, recalculation may be performed after the power increase is completed. The following considerations are noted:

1. For plants with technical specification requirements for gross specific radioactivity, chemistry personnel may wish to establish increased RCS sampling to monitor the Ar-41 concentration effect on this parameter.
2. For plants with technical specification requirements for dose equivalent Xe-133, Ar-41 has no effect on this parameter (unless, for some reason, the technical specification definition of dose equivalent Xe-133 includes Ar-41).

E.10 Pressurizer Steam Space Gas Control

Some plants periodically line up a pressurizer steam space sample line to the VCT in order to adjust/control hydrogen, or the buildup of other gases, such as helium. If such an operation were performed, chemistry personnel should periodically sample the RCS to monitor the effect on the Ar-41 concentration and consider the following:

1. Pressurizer steam space flow to the VCT may require periodic degassing of the VCT due to pressure increases.
2. Ar-40 in the pressurizer steam space could result in a gradual increase in Ar-41 in the RCS as the pressurizer steam space gas is introduced to the VCT gas space.

E.11 RCS Leaks

Leakage of gas from the RCS impacts dissolved gas parameters such as hydrogen and radiogases. Control of Ar-41 concentration may change if system leak rate occurs, such as:

1. Pressurizer relief valve leakage
2. VCT pressure relief valve leakage

Personnel may find that Ar-41 concentration decreases more quickly than usual. This results in the need for more frequent additions of argon gas to the RCS. Control of Ar-41 is normally still relatively simple in these situations. However, personnel need to be aware of this possibility. Chemistry personnel may be able to help provide early warning to operators and maintenance personnel that some kind of system change has occurred.

E.12 Example Ar-41 Trend

Figure E-6 shows a typical Ar-41 concentration increase upon initiation of argon injection and Figure E-7 shows a typical Ar-41 concentration increase during a routine argon gas injection.

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Figure E-6
RCS Ar-41 Increase Upon Initiation of Argon Gas Injection

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Figure E-7
Typical Ar-41 Concentration Increase During Maintenance Injection

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